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REACTORS

PROGRESS REPORT

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TECHNICAL DIVISION

M. D. Peterson, Director

REACTOR TECHNOLOGY

PROGRESS REPORT FOR QUARTER ENDING MAY 31, 1949

J. A. Lane and C. E. Winters

Associate Directors

EDITED BY

E. C. MILLER

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TECHNICAL DIVISION

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J. A. Lane and C. E. Winters
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JUL 18 1949

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ORNL 267
Reactors
Progress Report 5

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S U M M A R Y

MATERIALS TESTING REACTOR PROGRAM

Reactor Mock-Up. The erection and assembly of the reactor mock-up have been completed and the system has been placed in operation. No major difficulties developed. Minor adjustments are now being made, and preparations for the hydraulic and other operating tests are nearing completion. After the characteristics and reliability of the system have been established by operation over a period of time, some experimental modifications will be made, largely for purposes of simplification of structure and operation.

Reactor Design. Required experimental facilities and certain physical dimensions of the reactor structure have been well established. As a consequence the basic designs of the reactor structure and associated facilities have progressed rapidly. Specific design problems now being studied include shielding, reactor control devices, graphite ball discharge, and service facilities.

Corrosion. Autoclave testing shows promise as an accelerated test to predict qualitatively the susceptibility of beryllium to certain types of corrosion.

Corrosion tests have been initiated on QM (a Brush Beryllium Company sintered powder product) beryllium to compare it with cast and extruded metal as a potential reflector material. Results to date are encouraging but not conclusive.

Standard corrosion tests in simulated reactor cooling water indicate that the U-Al alloy core material of the aluminum-clad fuel assemblies will not be subject to severe preferential attack in the event of accidental exposure.

Titanium and zirconium stand up well under specified conditions of exposure to simulated reactor cooling water.

Metallurgical Engineering Development. Metallurgical work on the development and fabrication of fuel rods and beryllium reflector sections for the reactor has been transferred to the Metallurgy Division.

CONTROL AND REMOVAL OF AIR-BORNE PARTICLES

The methods of handling activity-bearing gases have been standardized on the basis of a classification of exhaust gas lines according to source, activity level, and chemical content.

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A summary is presented of facilities which have been installed for the treatment of off-gas lines from various sources.

Plans have been prepared for the installation of certain new and permanent air handling and decontamination facilities.

CHEMICAL ENGINEERING RESEARCH

The first phase of the liquid metal heat transfer study has been published as ORNL 361.

The results of the coolant survey will be issued shortly as ORNL 360.

Qualitative results and photographs of radiation stability tests on a great variety of plastics are presented.

Laboratory work has been completed on the investigation of the relation between the physical properties of solvents and mass transfer in a liquid-liquid dropwise extraction column. A report is in preparation.

ENGINEERING DESIGN AND DEVELOPMENT

A duplex-bellows, Flex-O-Pulse timer controlled, solenoid valve type pump for handling radioactive chemical solutions has been built but not yet tested.

A chain conveyor type isotope loader is being built for the ORNL reactor.

An experimental safety-shim control rod for the ORNL graphite reactor has been designed.

A corrosion program has been initiated to select suitable materials for use in a proposed homogeneous reactor using aqueous uranyl sulfate solutions.

Building 9204-1 at the Y-12 site is being modified to house the activities of the Engineering Development, Engineering Materials, and Engineering Research Sections of the Technical Division.

SHIELDING

The lid tank attenuation test facility for the ORNL reactor has been completed. A motor driven servo operation has been installed to increase the speed of obtaining gamma transmission data in the tank.

CHEMICAL PROCESS DEVELOPMENT

The progress of the Technical Division in the field of chemical process development during this past quarter is reported separately as ORNL 268.

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M A T E R I A L S   T E S T I N G   R E A C T O R   P R O J E C T

D E S I G N   P R O G R E S S

REACTOR MOCK-UP

During this period, design and procurement of all parts of the mock-up were completed.

Aluminum Tank. The aluminum reactor tank (shown in another section of the report in Figs. 15, 16, and 17) was shipped after a satisfactory pressure test at 80 lb per sq in. In attempting to make a light fillet weld to close all crevices inside this tank, it was found that the smallest weld attainable with existing equipment was 5/16", which in some cases caused serious interference with the reflector parts. Alternate methods proposed for closing these places were either peening or tamping soft 2S aluminum rod into the crevices. It was felt that neither of these alternates would entirely eliminate the possibility of water collecting inside the spaces, and that an open crevice would be preferable in that it would permit better circulation of the water. All thimbles for the mock-up tank are welded at the outside of the tank wall and one of these has an additional weld around the inside. Corrosion effects will be studied in the mock-up to aid in design of the pile tank.

Top Plug Assembly. At the close of the last quarter, the fabrication of of the top plug group was complete with the exception of the plug itself and the guide ring. The plug was held up pending receipt of the 2-1/4" × 80" diameter stainless steel plate. The guide ring was being held for heat treatment along with the top plug prior to final machining. About March 15, the large plate was received and the fabrication of the weldment was complete in early April.

On April 5, the two weldments (plug and guide ring) were shipped to Combustion Engineering Company for heat treatment and quenching. It was feared that due to the bulk, mass, and variation of plate thicknesses on weldments, serious distortion and damage to welds might result from quenching in water from 1900°F. However, the warpage on either part was negligible, and only minor weld damage to the top plug was observed. These welds were repaired after return of the parts to Y-12 and the plug showed no leaks through a soap film when tested at 75 psi air pressure. After final machining of the weld-

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ments, a partial assembly of the group was made at Y-12, including incorporation of the driving elements of a shim and regulating rod. Final assembly was made on the mock-up site.

Monitor Tube Design. The monitor tube consists of a modified pitot tube and thermocouple adapted as an integral unit with provision for conversion of the pitot tube pressure difference to a proportional electrical signal. This latter is accomplished by means of a compact differential transformer, the output of which is proportional to the movement of the soft iron core which is attached to a bellows. The movement of the bellows in turn is a function of the velocity head of the flowing water at the point measured. (See Fig. 1). A sample of this tube has been constructed and provisions are being made to calibrate the instrument prior to installation in the mock-up.

Reactor Control Console Mock-up. A reactor control console mock-up (Fig. 2) was designed and built with the following purposes in mind: To provide a central control position for the mock-up reactor; to check the practicability of the location of control switches and indicators for most efficient and safe reactor operation, and for possible use in the training of future operators of the Materials Testing Reactor.

The control console was designed so that all reactor controls could be reached by an operator seated in front of the console. The positioning of all control switches is such that all switches that must be operated separately or simultaneously can be reached with a minimum of effort on the part of the operator. The height of the console is such that instrument panels in the room can be readily observed by the operator without his having to leave his seat. A small desk area and drawer space have been provided for any recording or or other paper work that might be necessary on the part of the operator.

The photograph, (Fig. 2) shows the console as it now exists in the mock-up reactor building at ORNL. The console section to the extreme left of the operator contains adequate space for the eight shim rod control switches, position indicators, and various signal lights, also space for a mimic bus layout with its necessary signal lights. The console section, just to the left of the central section, provides for two regulating rod position indicators, a regulating rod selector switch, and necessary signal lights. The central section provides for two 50 cm galvanometer scales. The section just to the right of center provides for a scram switch, a speed selector switch, a shim-rod selector switch, a reverse by-pass switch, and all necessary associated signal lights. The console section on the extreme right of the operator is now being used for control switches and indicators for continuous testing of the reactor mock-up,

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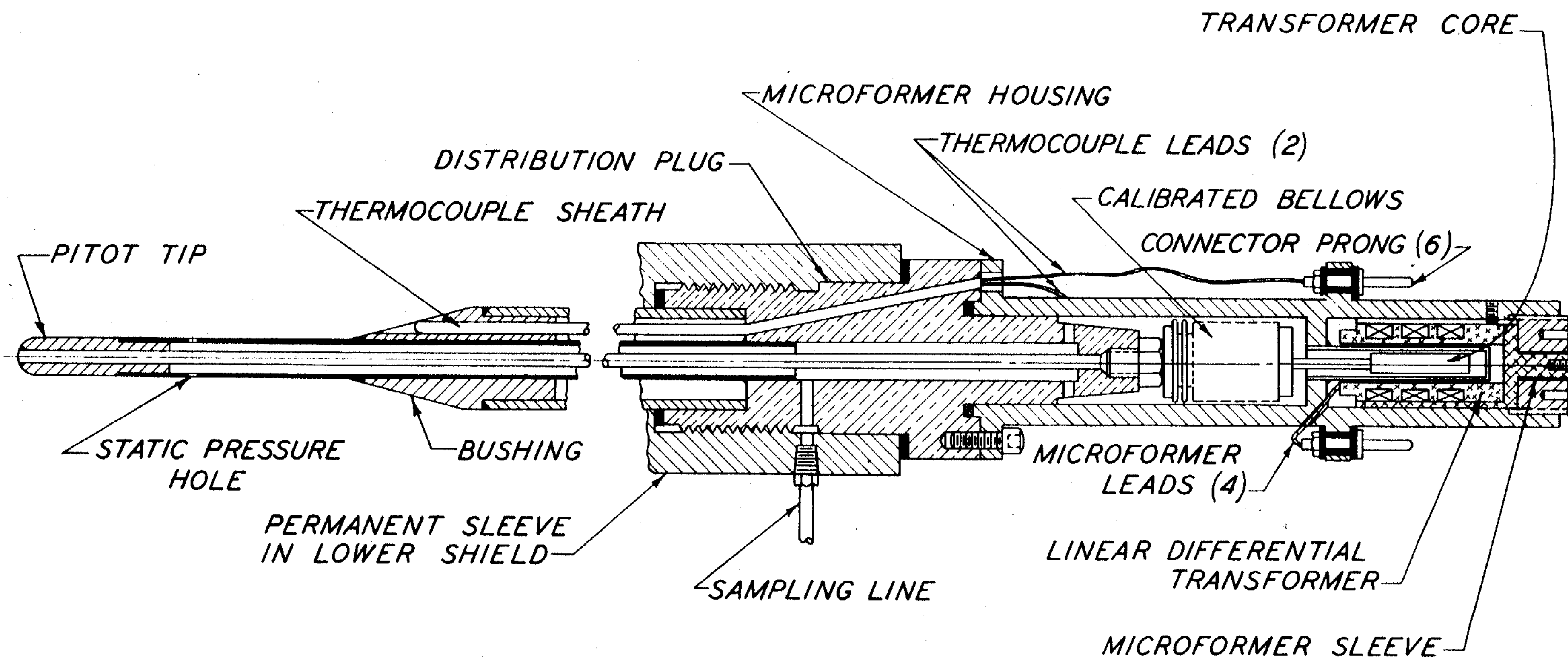

FIG. I

CROSS-SECTIONAL VIEW OF MONITOR TUBE

FIG. 2
CONSOLE TYPE CONTROL BOARD FOR PILE MOCK-UP

[redacted]

but for the Materials Testing Reactor, it will be used for controls of secondary importance.

COST ESTIMATE

Members of the Design Section have examined, in as much detail as possible, the specifications and requirements of the Materials Testing Reactor and have attempted to prepare a more realistic cost estimate than hitherto available. This estimate includes the reactor building, the reactor, and its cooling system, based on design information available as of April 4, 1949. It does not include the cost of development and operation of the site. The cost data for this estimate are from actual cost of parts of the mock-up, Brookhaven costs for graphite, thermal shield and biological shield, equipment price quotations from many manufacturers, and building cost data from ORNL and Y-12 Engineering Departments, AEC, and J. A. Jones engineers.

The following table summarizes the cost of major items and allowances for design, construction, and contingencies:

Summary of Cost Estimate for Material Testing Reactor

Materials Testing Reactor	$3,709,000
Reactor Building	2,907,000
Process Water System (Including retention basin and 3,000,000 gallon reservoir)	2,905,000
Reactor Air System	332,000
TOTAL DIRECT COST	$9,853,000
Design and Construction Drawings and Spec. at 20% of Direct Cost	1,960,000
Inspection at 1½%	147,000
	$11,960,000
Design and Construction Fees at 5%	600,000
Contingencies at 20%	2,400,000
	$14,960,000

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EXPERIMENTAL FACILITIES

A second set of study layouts was completed on the experimental facilities late in February. Several changes were introduced in ensuing meetings within the Laboratory, and at a meeting of the Reactor Steering Committee on March 22, the facilities were given verbal endorsement with minor modifications in the proposals. The facilities were formally submitted to the committee April 6.

The presently proposed facilities are the same as described in ORNL 323, Part I with the exception of the following changes approved by the Steering Committee:

1. *There are now six 2 inch down-beam holes instead of four as planned previously.*
2. *Instead of two 5 by 5 inch facilities through the reactor tank adjacent to the active section, only one is now provided.*
3. *The number of vertical graphite holes is now firmly established as follows: six vertical instrument holes, 2½ inch diameter, from the top of the reactor structure to the side of the reactor tank at the pile centerline; two 2½ inch vertical holes through the graphite to the canal; four 4 inch, thirteen 2½ inch, and thirty-nine 2 inch diameter holes from the top of the reactor structure to points below the pile centerline in the graphite.*
4. *Instead of a thermal column on the north side of the reactor, a shielding facility is provided on the north side, with the thermal column shifted to the south side of the reactor. Both facilities are fronted by bismuth windows at the thermal shield. In connection with the shielding facility, a converter plate is provided with access from the sub-reactor room. This facility is described in more detail below.*

A model of the pile structure and experimental facilities was completed in late April. Figures 3-7 show the experimental and service facilities in relation to the pile structure as a whole.

Shielding Facility. Preliminary layouts of a facility for the testing of shielding materials in the Materials Development Reactor have been prepared. They are based on information presently available from Oak Ridge National Laboratory and Argonne National Laboratory drawings of the reactor and accompanying building designs, and on information supplied by the Shielding Group of ORNL.

The purpose of the layouts is to provide a facility for the testing of possible shielding materials for future reactors, under conditions of thermal and fast neutron fluxes simulating as closely as possible the values of the

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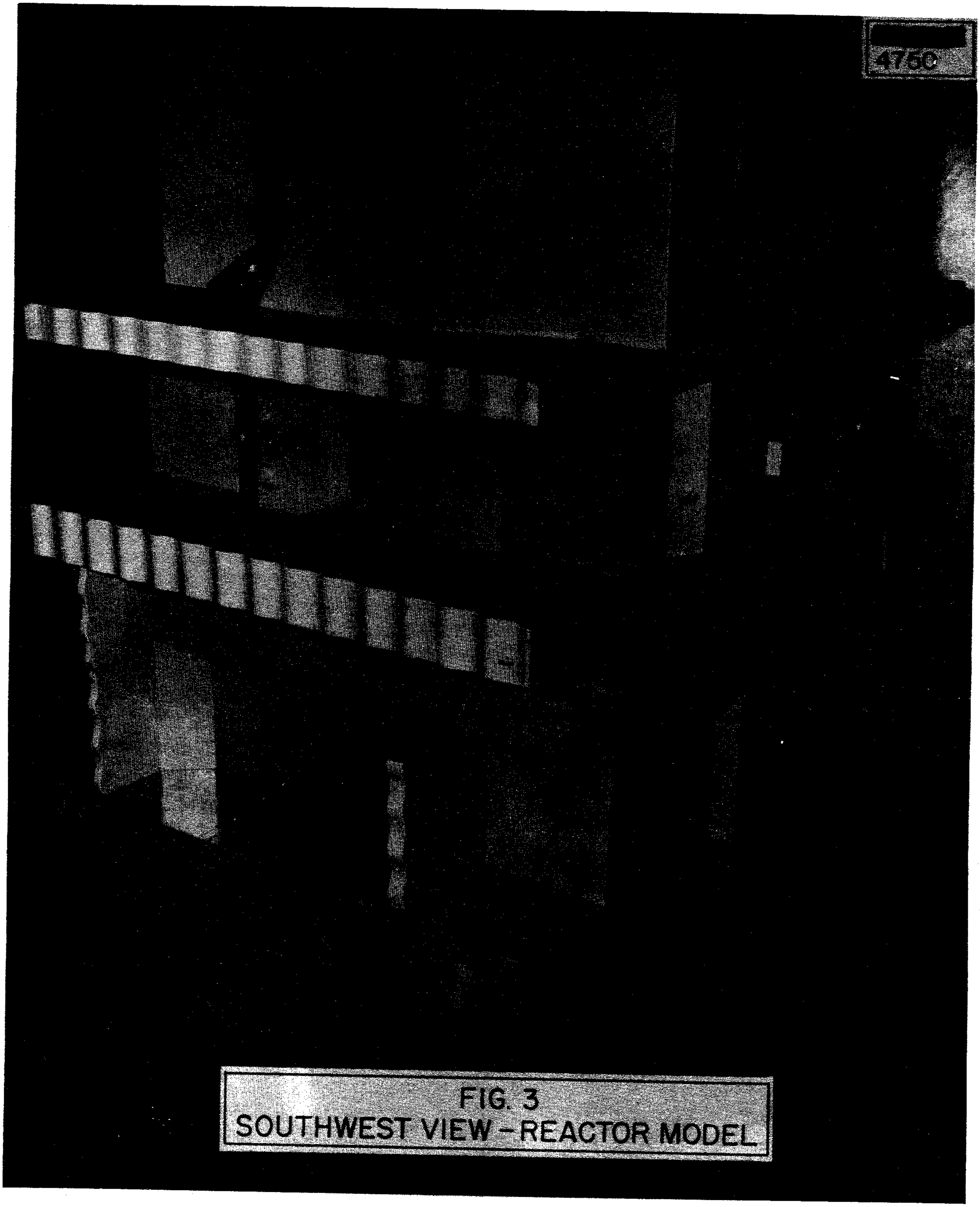

FIG. 3
SOUTHWEST VIEW – REACTOR MODEL

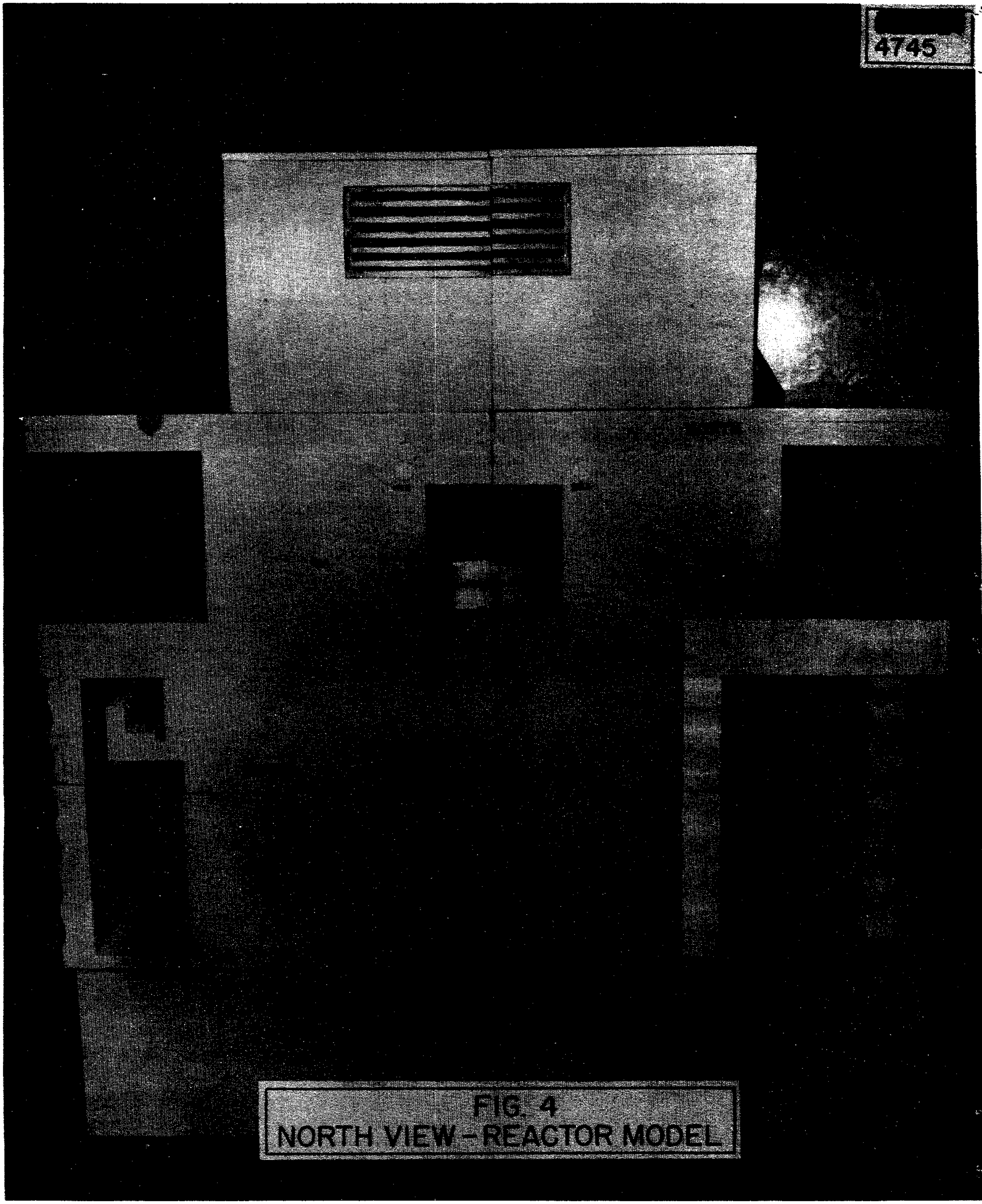

FIG. 4
NORTH VIEW – REACTOR MODEL

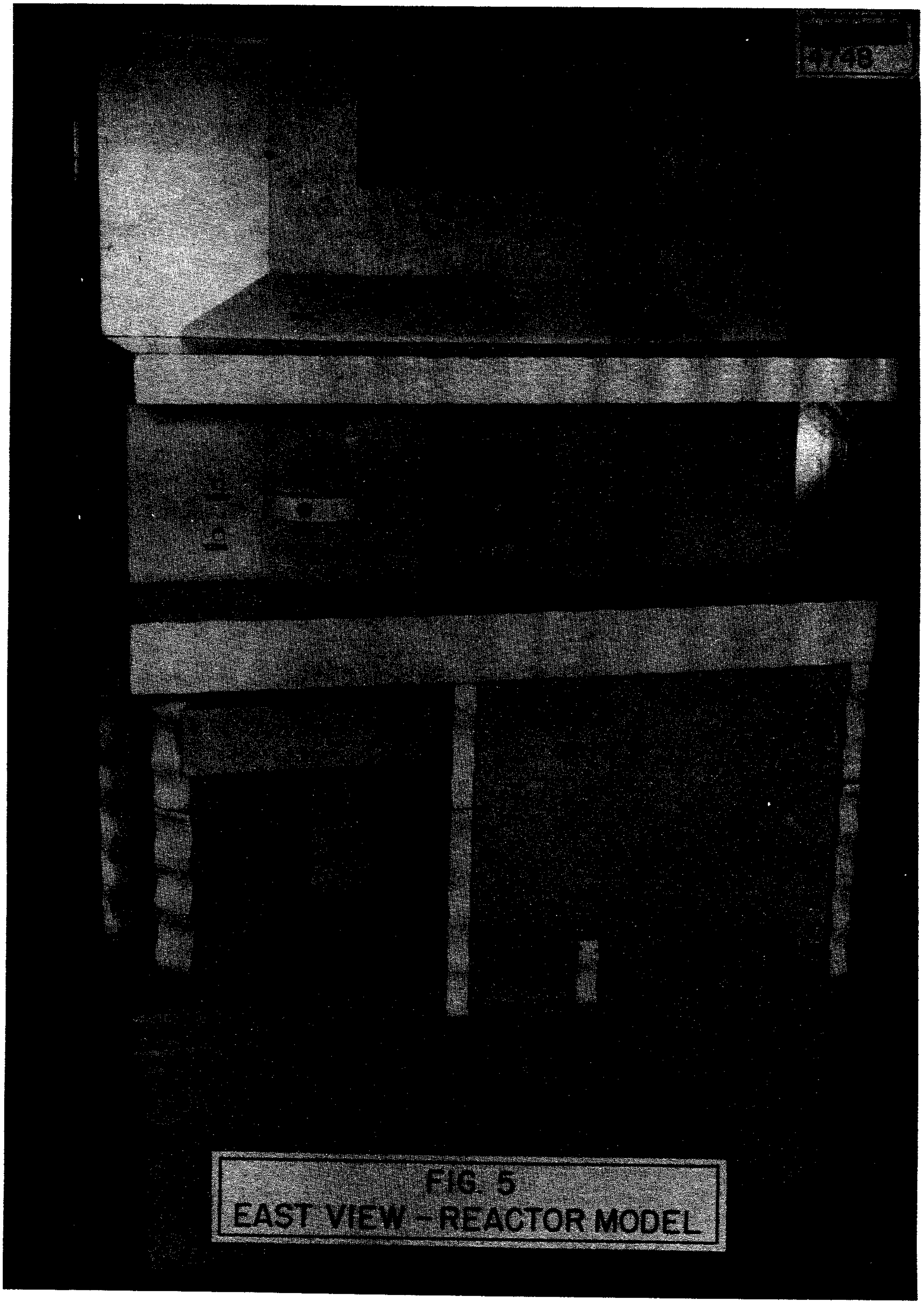

FIG. 5
EAST VIEW – REACTOR MODEL

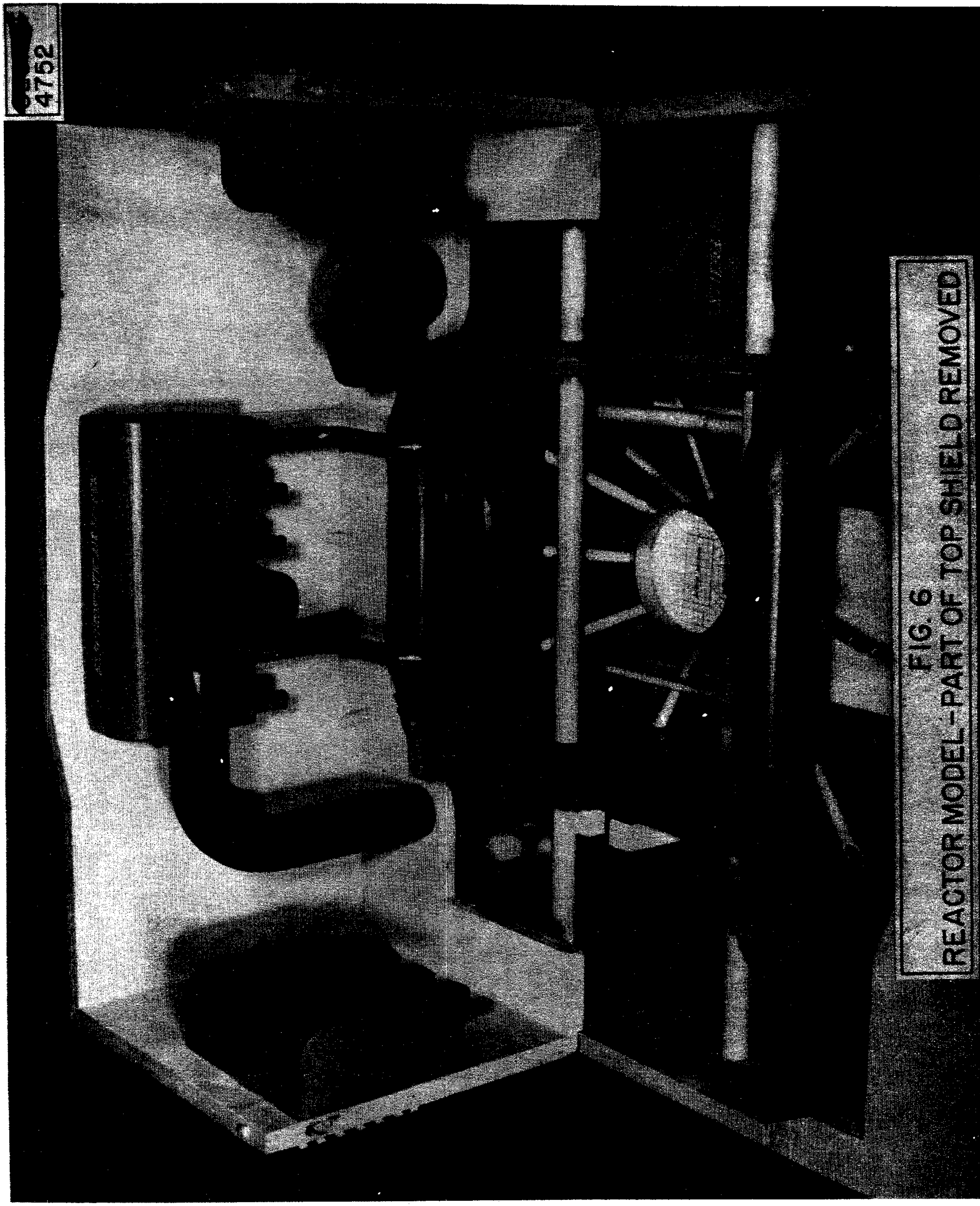

FIG. 6
REACTOR MODEL – PART OF TOP SHIELD REMOVED

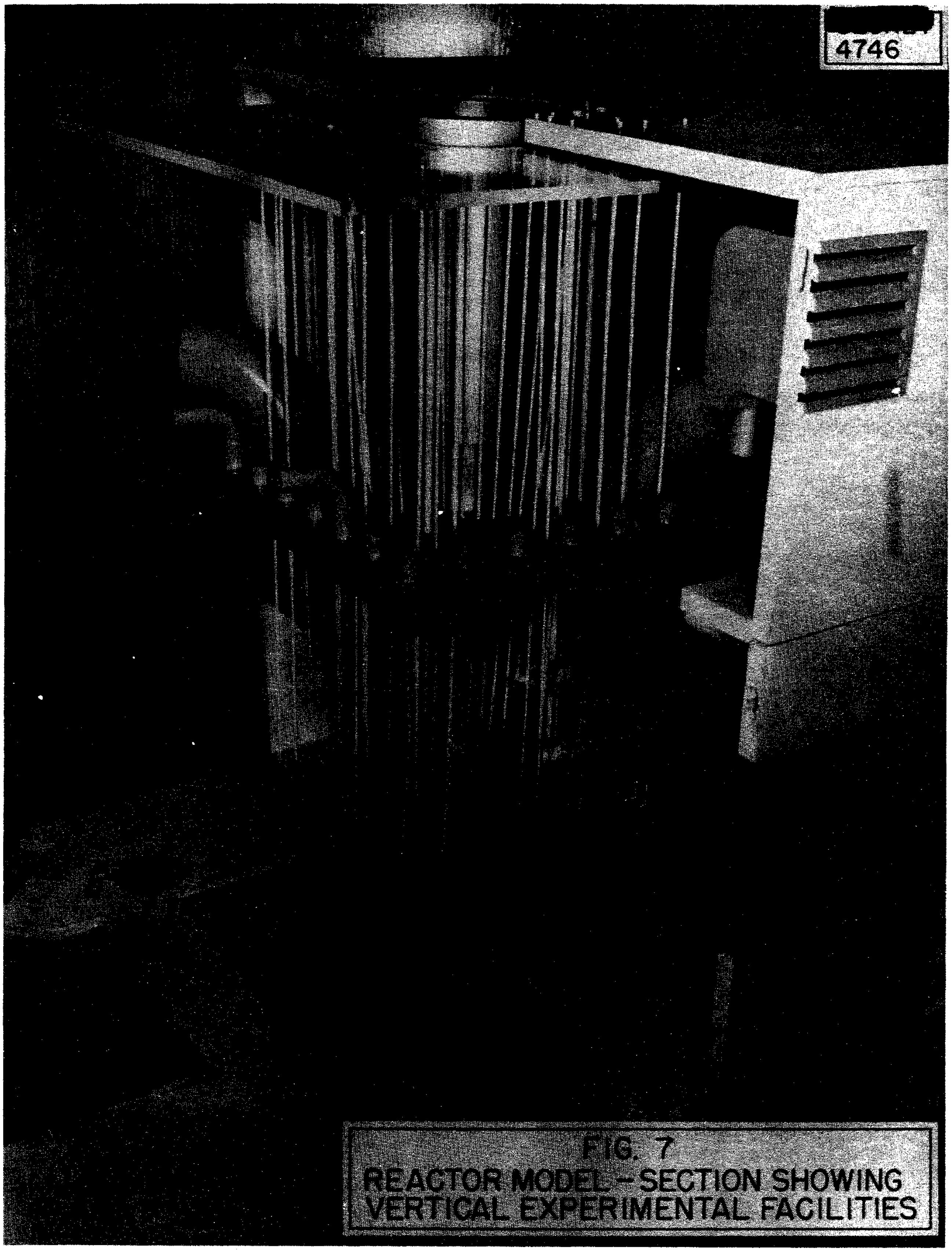

FIG. 7
REACTOR MODEL – SECTION SHOWING
VERTICAL EXPERIMENTAL FACILITIES

fluxes expected in these reactors. The design values of the thermal and fast neutron fluxes in the Materials Testing Reactor are of the order of magnitude desired.

Requirements of the facility are as follows:

1. *The shielding material specimen slabs shall be capable of being changed during pile operation without disturbance to the operating schedule.*
2. *The slabs shall be placed as close to the reactor core as possible to permit exposure to as high a flux as possible.*
3. *The slabs shall completely fill the square opening through the reactor shielding and shall be equipped with remote handling attachment devices.*
4. *Adequate provision shall be made for safely storing those portions of the specimen which will become excessively radioactive.*
5. *The slow neutron source from the reactor shall be changed to fast neutrons by means of a converter.*
6. *The converter shall be movable relative to the test slabs, in order to simulate an infinite plane source of neutrons.*
7. *The neutron converter shall be placed in the reactor in such a manner as to be remotely discharged.*
8. *The test hole through the reactor shielding shall be lined with a material such that back-scattering of neutrons, with consequent distortion of experimental data, shall be minimized.*

Briefly the facility shown in the layouts consists of a rectangular water tank fastened to the face of the reactor, in which a dolly is moved in and out of an opening in the reactor shield by means of a motor-operated screw. Shield specimens may thus be inserted and removed under water. A deep well for storage of active specimens is provided at the outer side of the tank.

REACTOR STRUCTURE

As of this writing, the physical dimensions of the reactor structure are well established. Approval of the experimental facilities allowed more detailed study, and drawings are completed showing the basic dimensions of the structure above the first floor level and a simplified vertical section of the reactor tank with a portion of the pile structure, on which the basic dimensions of the tank assembly, graphite reflector, thermal shield, bottom plug, top plug, etc., are set up for reference purposes. These constitute the premises upon which future work will be based.

Additional drawings are in process which will establish the basic dimensions of the structure from the canal to the top of the structure and in the

[redacted]

sub-pile room. It is the purpose of these drawings to establish the requirements of the experimental facilities and the main service facilities in the basement and sub-pile room. It is hoped that this information will be useful to the Argonne Group in designing the basement in detail.

Studies have been initiated on the primary service facilities within the structure. These include reactor cooling water, graphite cooling air, shutdown overflow lines, and safety vent-purge lines. These facilities are shown in Figs. 6-7 of this report. The information developed on these facilities is as follows:

Graphite Cooling Air. As was pointed out in the early meetings of the Steering Committee, it was felt desirable from the standpoint of safety to have the air introduced and exhausted from the top of the thermal shield. Such an arrangement allows the thermal shield to act as a *secondary tank,* in that it will leak only small quantities of water should the reactor tank rupture. The probability of ever having a dry reactor is thus reduced.

The air system shown in Figs. 6-7 is based on an estimated requirement of 25,000 cfm. Air is taken directly from the reactor building through glass wool filters (electrostatic filters shown in model photographs) located on the four faces of the reactor structure near the top level. From the distributing chamber immediately behind the filters, the air is drawn through thirty-one 8 inch diameter ducts into distributing ducts around the upper periphery of the thermal shield. The air is then conducted downward through the thermal shield annulus, thence upward through the graphite cooling passages into a plenum chamber above the graphite, from which the air enters two 36 inch discharge ducts. The diametrically opposed discharge ducts open upward and make 180-degree turns to exit from the reactor structure vertically downward through the basement.

Specifications for the air system within the pile structure are as follows:

Volume: 25,000 cfm
Inlet ducts: 8 inch diameter, 31 total
Air velocity in inlet ducts: 2250 fpm
Discharge ducts: 36 inch diameter, 2 total
Air velocity in discharge ducts: 1770 fpm
Filters: AAF Airmat PL-24 (24 required)
Air velocity at filter face: 260 fpm

Each air inlet duct has three 90-degree turns to effect a radiation block from the thermal shield to the outside of the structure. The adequacy of this number of turns has been confirmed by calculation.

[redacted]

Inlet Water. The arrangement of the inlet water lines is shown in Fig. 6. From the basement, two 24 inch lines rise vertically through the reactor structure to a level about 16 feet 10 inches above the reactor centerline. The lines enter the reactor tank radially and are diametrically opposed to minimize the impact effect within the reactor tank.

Exit Water. The arrangement of the exit water lines is shown in Fig. 6. Each of the two 24 inch lines runs horizontally beneath the thermal shield to the nearest point where a 90° upward turn can be made through the concrete structure. The vertical risers turn 90° to become horizontal at an elevation 7 feet 4 inches above the pile centerline, cross over the opposite side of the structure to turn downward 90° to approach the basement. In effect, an inverted U-tube is inserted into the discharge waterlines. With proper venting to prevent siphoning, this arrangement provides a safety device, in that rupture of a discharge waterline outside the pile structure would not cause the tank to be run dry.

Vent and Purge Lines. Two 2 inch diameter vent-purge lines run from the top section of the reactor tank, just opposite the top plug, to the two exit waterline cross-overs in the pile structure. Their purpose is twofold: (1) To constantly purge the top of the reactor tank of gases, and (2) to prevent the exit cross-over from siphoning the reactor tank dry should they ever be required to fulfill their intended purpose as explained earlier. These lines are direct connections with no valves, thereby eliminating the necessity of automatic valves in conventional vent lines to the atmosphere, as was proposed earlier. These lines were not built into the model and hence are not to be seen on Figs. 6-7.

Shutdown Overflow Lines. Two 8 inch diameter overflow lines have been provided. These lines run from the top section of the reactor tank into the warm drain disposal in the basement, their purpose being simply to minimize the probability of running the reactor tank over while flowing water through the reactor after shutdown with the top plug removed. Each line will be provided with appropriate valves interlocked with the main water system in such a manner as to prevent start-up with the valves open. These lines were not included in the model.

THERMAL SHIELD DESIGN

The thermal shield design for the Materials Testing Reactor is now in the layout stage. Various methods of construction have been considered, such as laminating from 1 inch to 2 inch plates, casting, and the use of armor plate, but

[redacted]

after due consideration of each, a method of fabrication using 4 inch, 6 inch, and 8 inch rolled mild steel plate, approximately 5 to 6 feet wide and 12 feet long, is being followed. It is now proposed that these plates be welded in position during the erection of the reactor. The holes in the shield will be flame cut to a tolerance of plus or minus 1/32 of an inch (similar to tolerances held for the same type of work on the Brookhaven Reactor). There are a few problems now being considered concerning various methods of allowing for expansion in shield. The completed design of the thermal shield cannot be made until an acceptable experimental facility sleeve design has been completed, which in turn will determine the size of the various holes to go through the shield. Work is now in progress on the design of these sleeves.

NEUTRON ABSORBING CURTAIN

A curtain, commonly known as the *cadmium curtain,* will be used as a neutron absorbing shield between the thermal column and the graphite reflector and also between the shielding facility and the graphite reflector.

As presently proposed, the curtain will be installed to operate between the inner and outer walls of the thermal shield. The present conception locates it below the thermal shield and in a location that makes it possible to raise the curtain into closed position. It is being designed in such a way that maintenance probabilities have been cut to a minimum, but if it ever becomes necessary to remove the curtain it may be done by the removal of a plug into the sub-reactor room. Two materials are being considered as the neutron absorbing medium, cadmium and *Boral*, an aluminum-boron carbide mixture developed by the shielding group. A study report is now being prepared on this design.

CONTROL RODS

Two shim rods and one regulating rod complete with driving mechanisms, were completed and assembled on the top plug for the mock-up. It was found that double pole limit switches are required on the shim rod drive where the original design specified single pole. To adapt a double pole switch to the existing housing, a new bracket was designed which incorporates the feature of a cover enclosing the adjustment screw. A sectional view of this switch mounting is shown in Fig. 8.

Studies are being made of the shim rod design with the purpose of redesigning to the following objectives:

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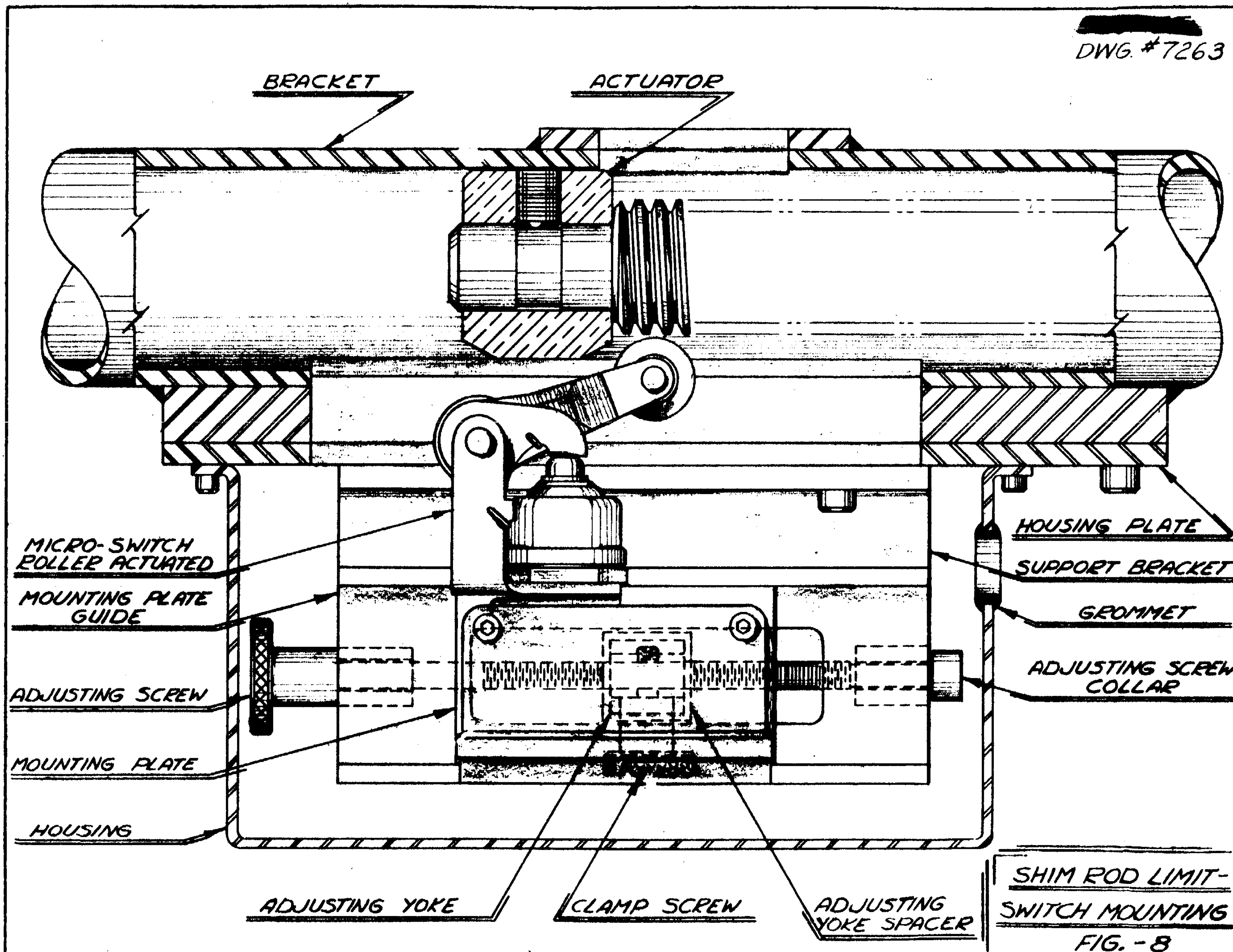

SHIM ROD LIMIT-SWITCH MOUNTING
FIG. -8

[redacted]

1. *Elimination of screw joints.*
2. *Simplification of fabrication for better economy.*
3. *Replacement of stainless steel with aluminum where feasible.*

Attention has been focused on the rod itself although the effects on the rod of possible changes in the dash pots, magnetic coupling, or bearings have been anticipated.

The changes presently under consideration for accomplishing these objectives are as follows:

1. *Use of extruded aluminum tubing in place of present stainless steel welded tubing, permitting use of welded joints at all intermediate points except at the bottom and top, where stainless steel and magnetic steel, respectively, would still be used.*
2. *Insertion of the cadmium into the rod in the form of an aluminum-cadmium sandwich, folded into box shape.*
3. *Joining the stainless steel shock section to the lower bearing section by threading it into an aluminum plug welded to the lower extruded section, then locking the threads with a taper pin driven in and peened in place.*

Feasibility of these changes is being checked by experiments.

BOTTOM SHIELD

Investigation has been made of the possibility of substituting lead as the shielding material to be used in the bottom of the reactor tank, instead of alternate layers of iron and water as originally proposed. Preliminary calculations and layouts indicate that this is feasible and results in reduced total thickness required in the bottom shield. In these layouts the bottom shield is removable into the sub-reactor room and hence into the canal. Removable lead plugs will be provided to shield against streaming through monitor tubes and shock absorber transmitter rods.

GRAPHITE BALL DISCHARGE

Provisions are being made to permit removal of the 240 cu ft of one inch diameter graphite balls used in the reflector. Present designs provide for dropping the balls into the basement through two 8 inch steel pipes with sufficient concrete for shielding. An enclosed truck loading station, truck and special control valves will be designed for removal of the balls.

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ASSEMBLY OF THE REACTOR MOCK-UP

The erection of the external structure, equipment, pumps, piping and control house of the reactor mock-up was completed before this quarter. During this past quarter, the reactor tank, the simulated active lattice, and the reflector have been assembled and the system put in operation.

The reactor tank comprises the top plug and six separate sections, A, B, C, D, E and F. The aluminum section D contains the active lattice and reflector. Water, at the rate of 15,000 gallons per minute, flows into tank section A, downward, and out through section E. The top plug contains the drive mechanisms for the various control elements and must be aligned very closely with the tank sections below it for the control rods to function properly. Therein lies the major problem in the assembly of the reactor.

Figures 9-22 show the assembly procedure, which may be briefly described as follows:

1. *Assembly and alignment of section D, containing the active lattice and reflector support castings.*

   (a) *The aluminum tank section D was mounted and accurately leveled on a supporting frame. The lower support casting was raised into section D and fastened in position (Fig. 9).*

   (b) *The shim rod bearing support casting was clamped in position underneath the lower support casting (Fig. 10).*

   (c) *The upper locking mechanism was then assembled, fastened to the shim-safety-rod upper bearing casting, the grid spacer, and the upper fuel grid (Fig. 11).*

   (d) *The upper support casting was clamped to the lugs at the top of section D and the assembly (shown in Fig. 11) lowered inside it. Figure 12 illustrates how preliminary alignment was obtained by inserting a shim and regulating rod through the assembled castings. Better alignment was accomplished with piano-wire plumb lines through the center of the rod bearing holes. The plumb was a heavy lead weight which was damped in a bucket of oil. All critical points were marked for doweling.*

2. *Tank sections E and F were positioned on the structural steel support in the sub-reactor room (Fig. 13).*

3. *Section A was lowered into position, shimmed, and bolted to the supporting structural steel, after leveling with a precision spirit level (Fig. 14).*

4. *After aligning and doweling the inner assemblies of section D, the tank was dismantled except for the lower support casting. Fig. 15 shows the tank being lowered through section A into position on section E. In Fig. 16 the tank is in position, and in section C, the bellows attached to it.*

5. *At this stage of the assembly the top plug was placed on section A, and sections D, E and F were shifted into alignment*

[redacted]

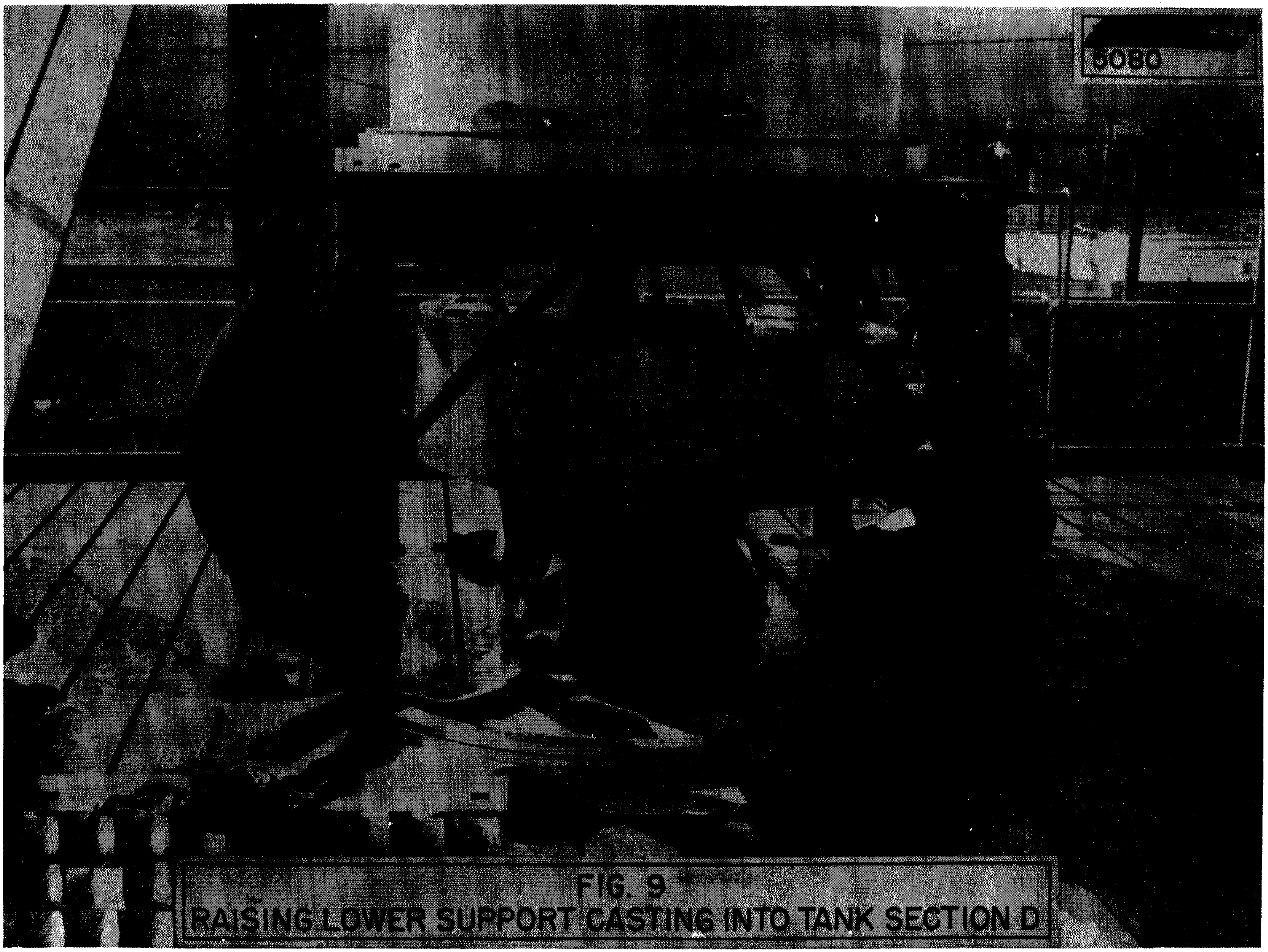


FIG. 9
RAISING LOWER SUPPORT CASTING INTO TANK SECTION D

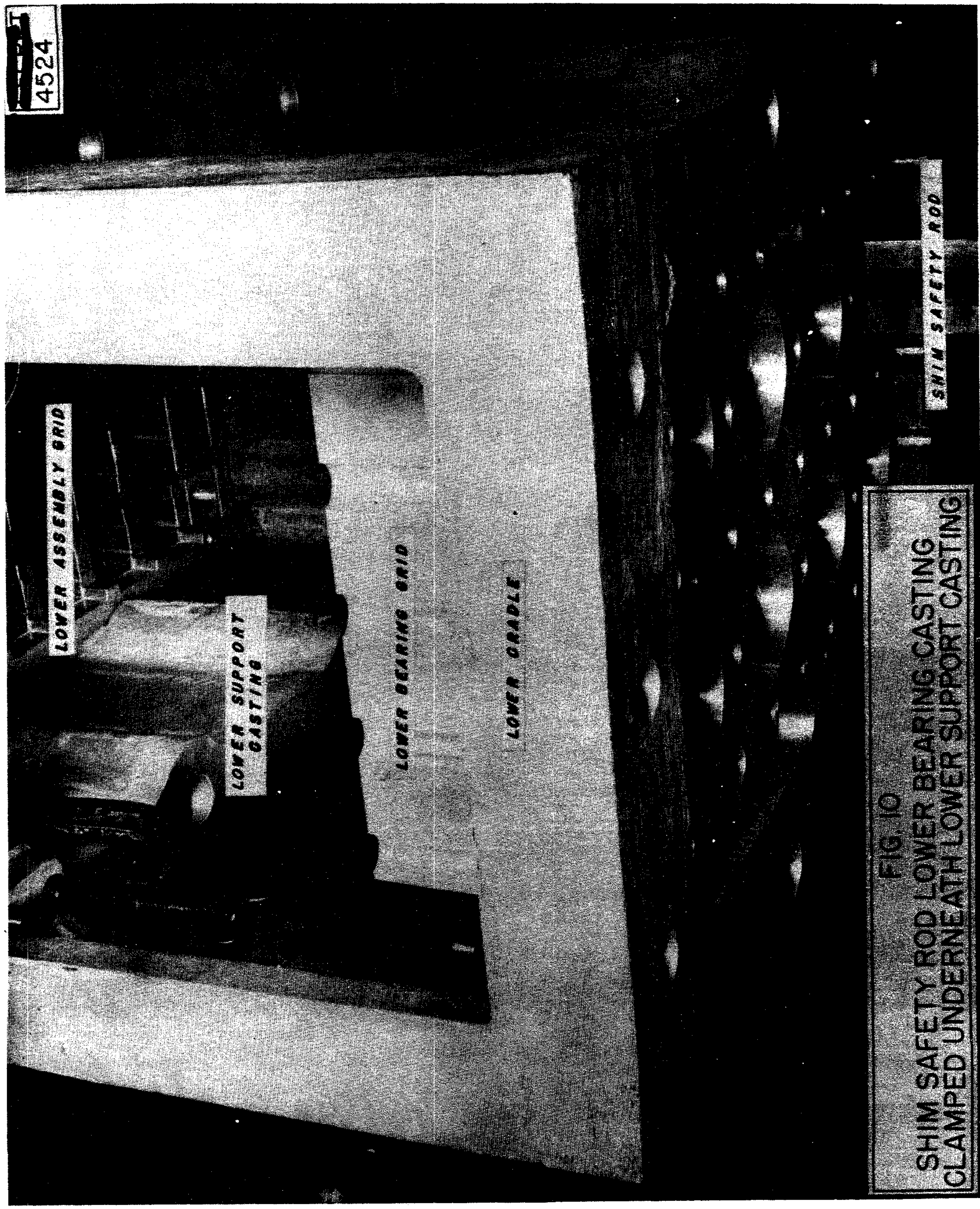


FIG. 10
SHIM SAFETY ROD LOWER BEARING CASTING
CLAMPED UNDERNEATH LOWER SUPPORT CASTING

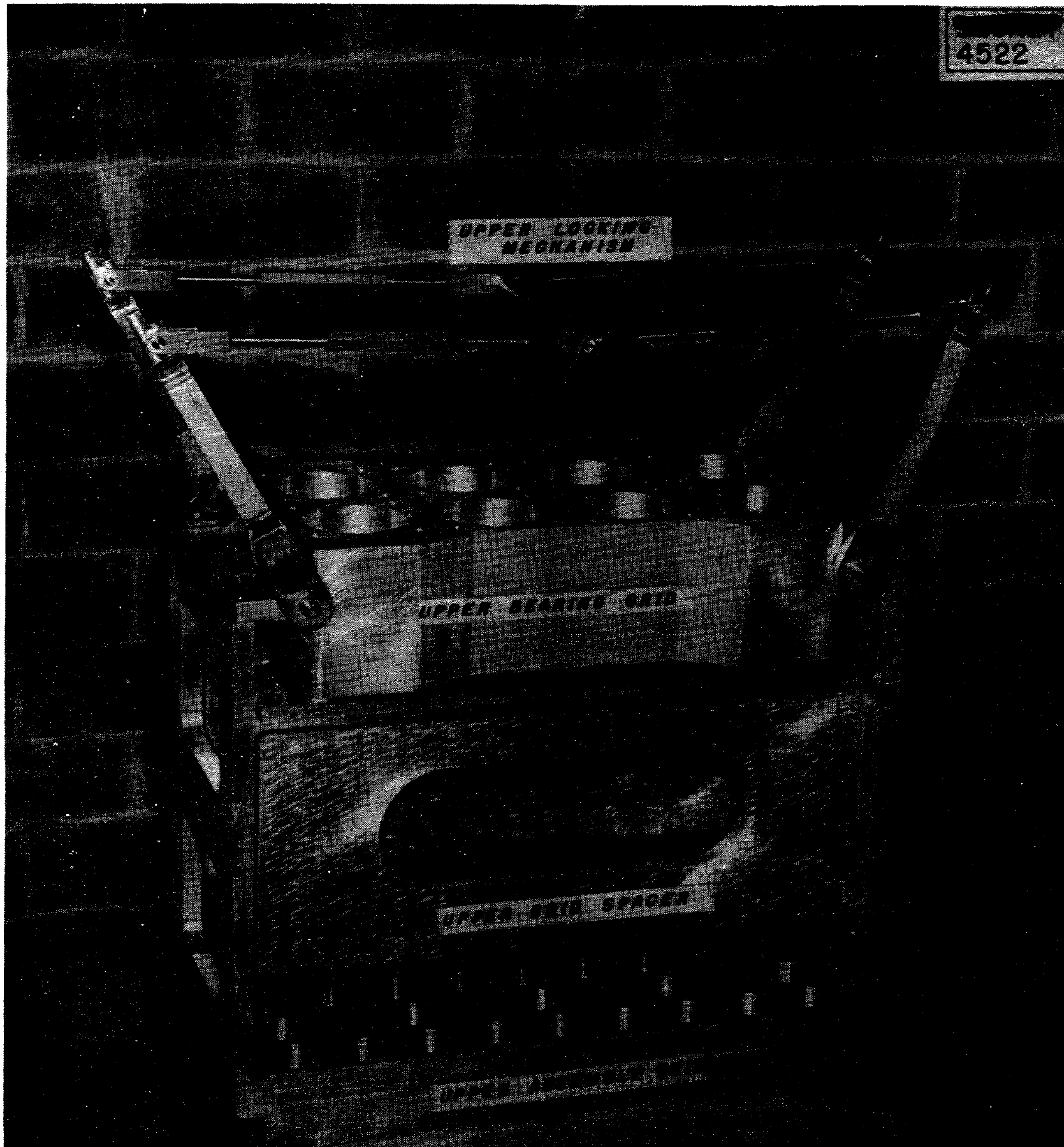
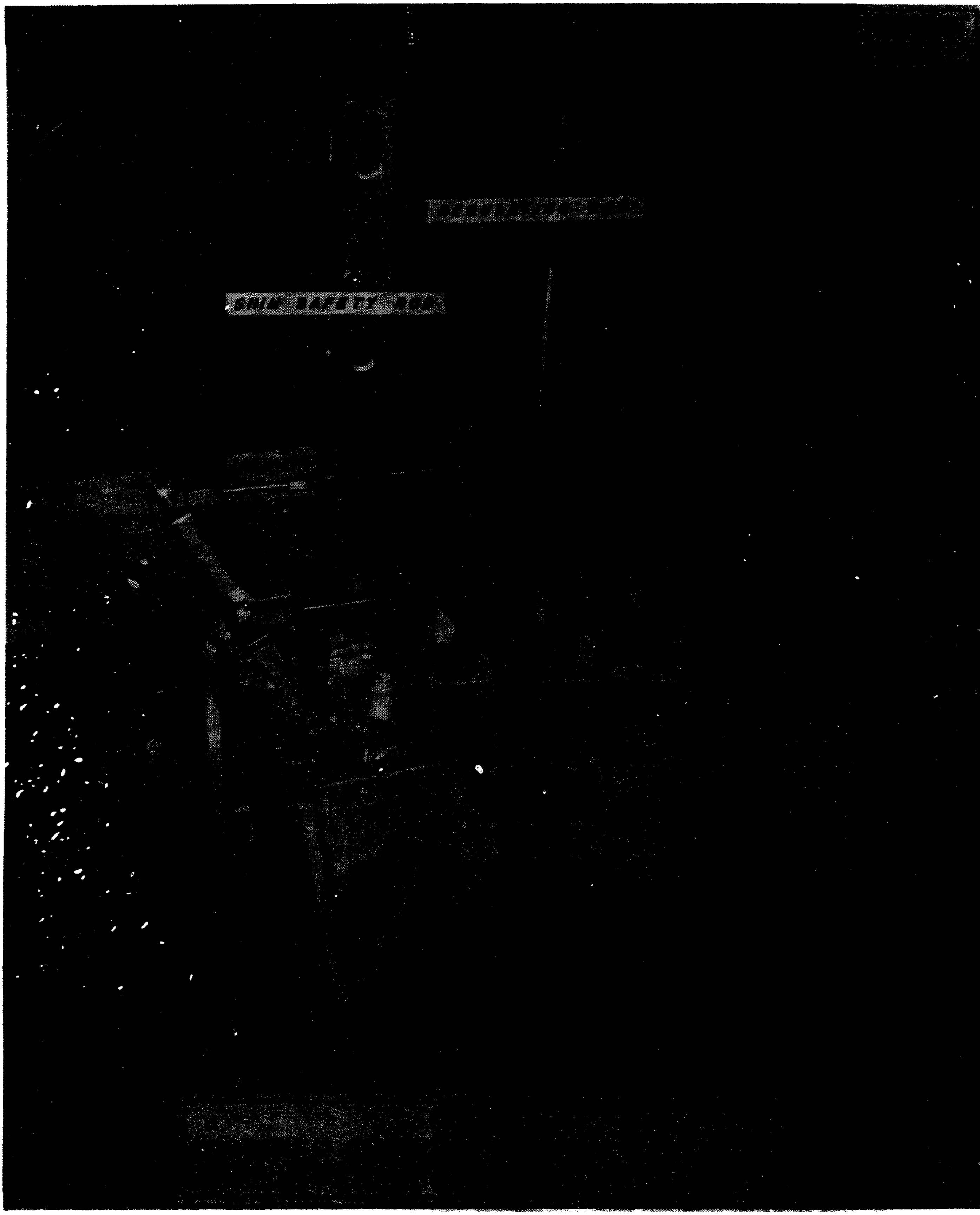


FIG. 11
ASSEMBLY OF THE UPPER LOCKING MECHANISM,
SHIM SAFETY ROD UPPER BEARING CASTING,
THE GRID SPACER, AND UPPER FUEL ASSEMBLY GRID


SHIM SAFETY ROD

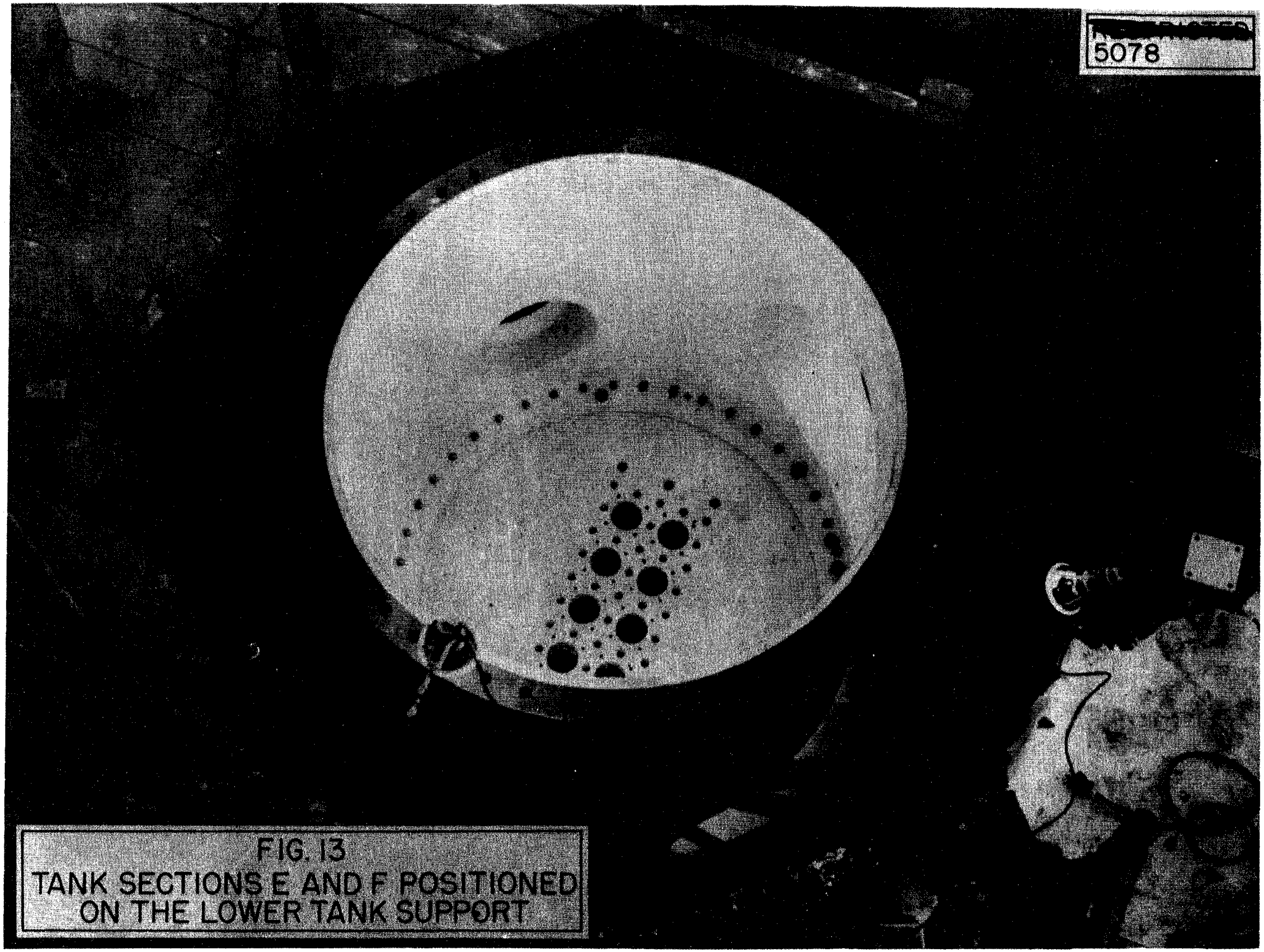
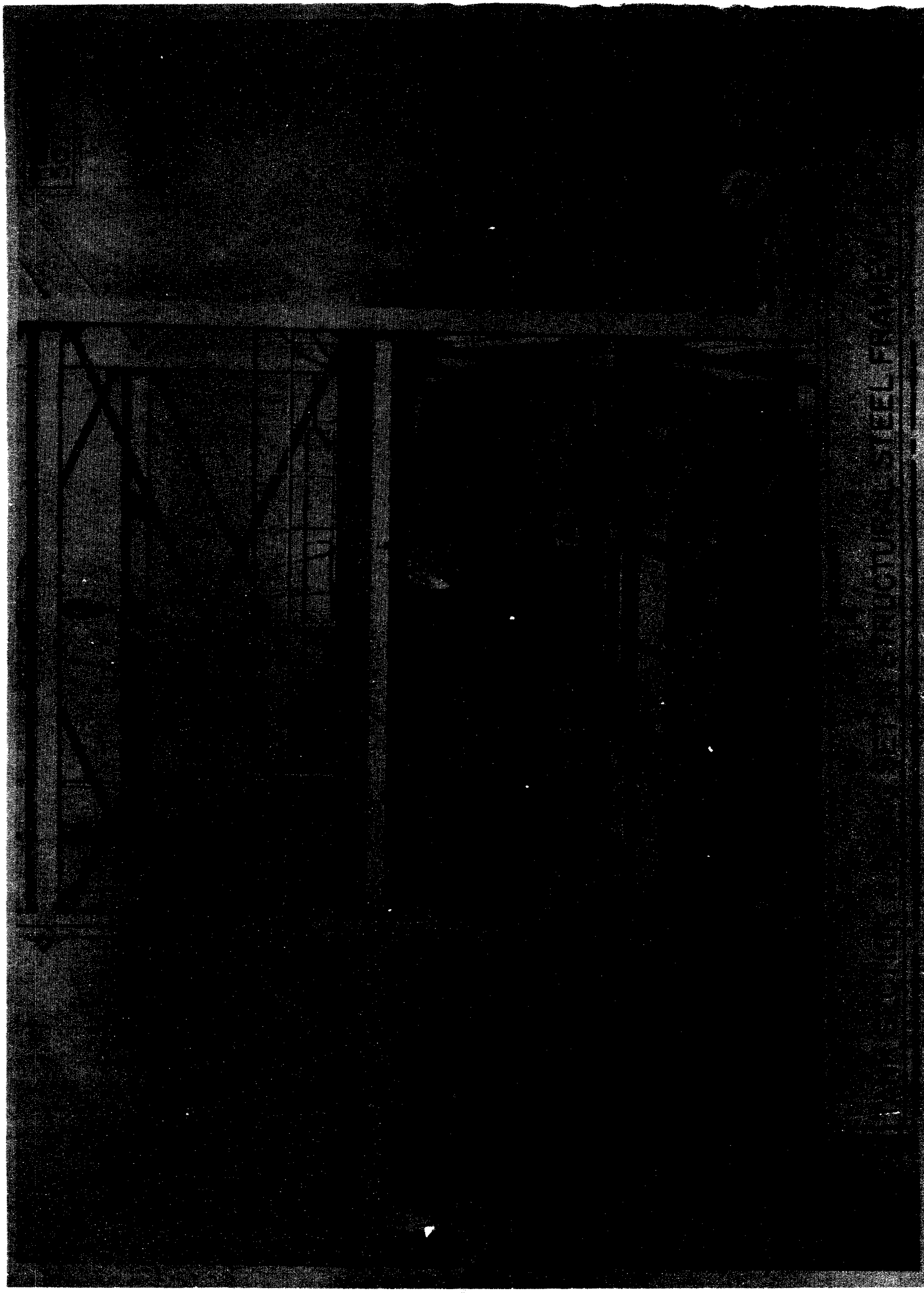


FIG. 13
TANK SECTIONS E AND F POSITIONED ON THE LOWER TANK SUPPORT



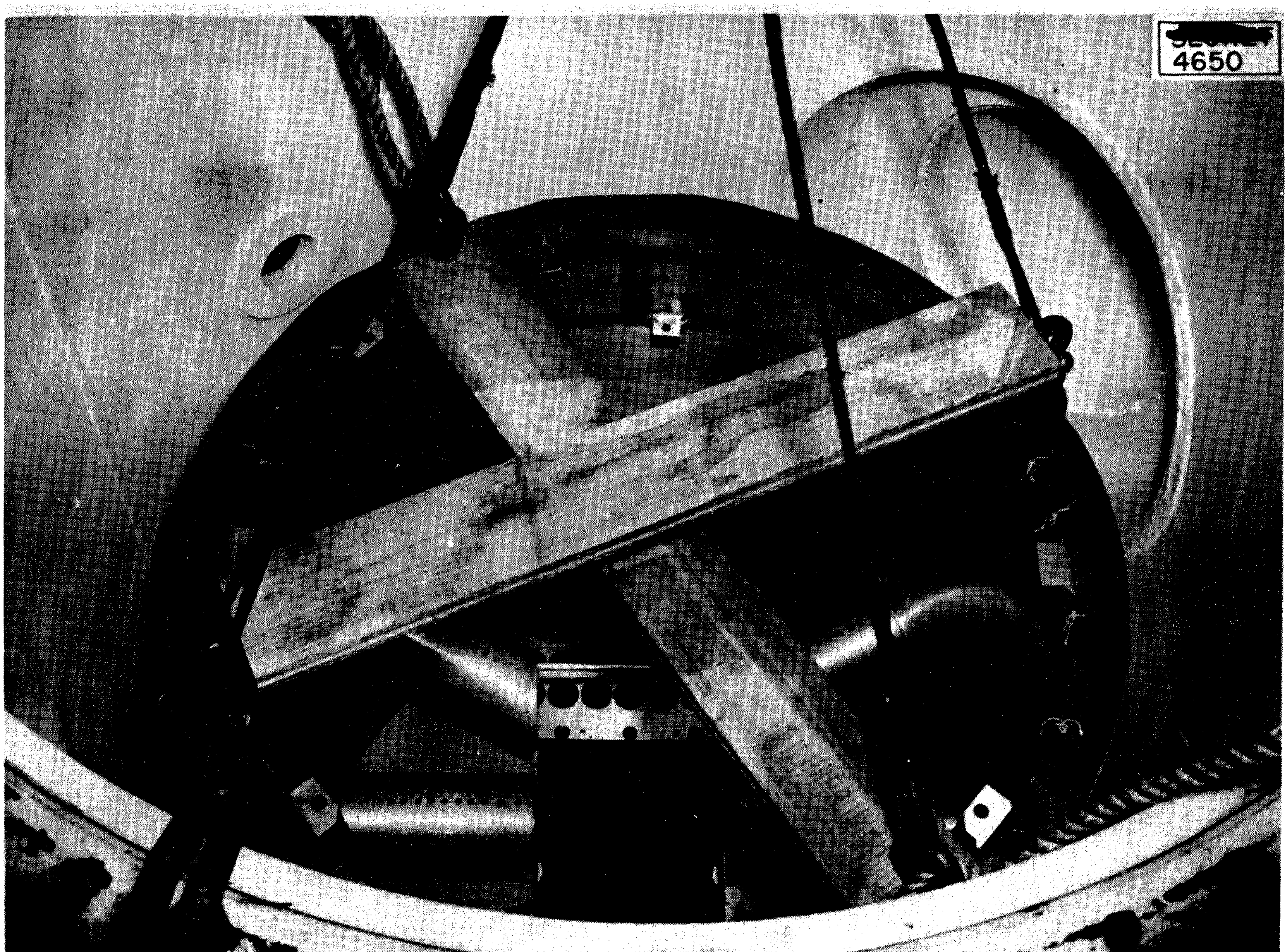


FIG. 15
SECTION D WITH LOWER SUPPORT CASTING BEING LOWERED THROUGH SECTION A

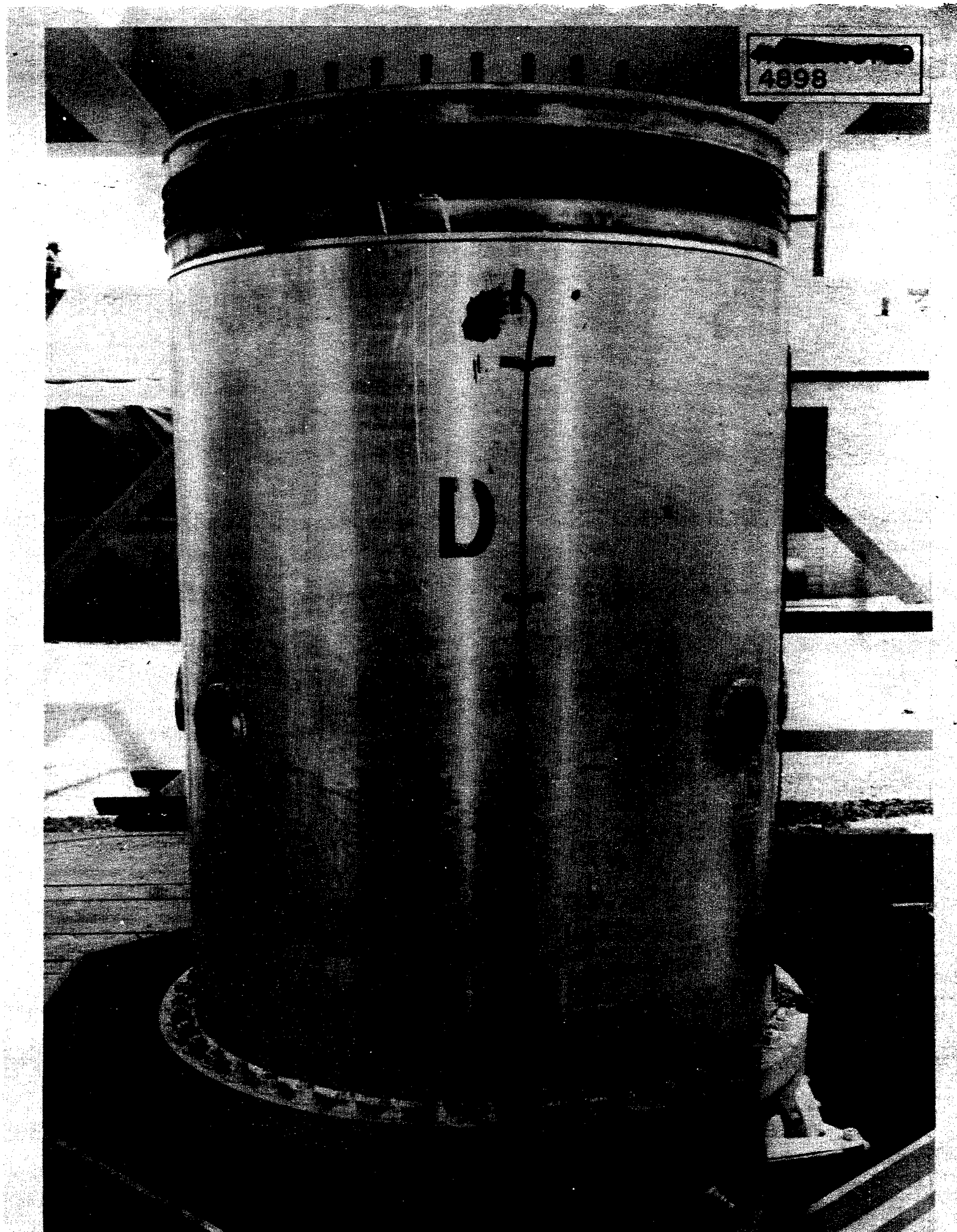


FIG. 16
ALUMINUM TANK SECTION D IN POSITION ON SECTIONS E AND F

with the top plug with jack bolts against the supporting plates on section F. The three sections were then leveled as indicated by a precision level set on the top flange of section D. Plumb bobs dropped from the top plug through the shim rod dash pots in section F indicated correct alignment and orientation. The 16 inch inlet and exit piping was connected and tightened with plumb lines still in position. Tank sections were doweled at this time.

6. *Assembly of the reflector portion of section D was begun. Figure 17 shows the partial and Fig. 18 the complete reflector assembly with the upper support casting in position.*
7. *The drive mechanism of two shim-safety rods and one regulating rod were then assembled on the top plug (Figs. 19-20).*
8. *With the assemblies of reflector and top plug completed, tank section B was set in place with the spider support ring on its upper flange (Fig. 21). After alignment of the spider support ring with the top plug and D and F sections, section B and the spider support were doweled in position. The completed tank assembly is shown in Fig. 22.*
9. *The lower grid (Fig. 17) of the active lattice was then loaded with two shim rods and 19 dummy fuel assemblies, and the remaining holes filled with mock reflector assemblies. Then, with the upper grids and the top plug in place, the regulating rod was attached to its drive mechanism and the positions of shim-safety rod magnets set. Position indicators were adjusted to zero readings and control system plugged into connectors from the control operating room. After preliminary operation of the control system, the pumps were started and water circulated at a rate of ~ 12,000 gpm.*

The entire system was put in operation within 30 days after assembly was begun. No major difficulties appeared in the fitting together of the various parts. However, a number of minor adjustments were necessary and many desirable changes have become evident.

GASKETS FOR USE IN MOCK-UP REACTOR TANK

The gasketing of the various joints between sections of the mock-up tank became a problem when several of the half inch wide, flat 3-S aluminum gaskets specified for use failed to seal. For initial demonstration of the mock-up it was necessary to replace these particular gaskets with Neoprene gaskets which held satisfactorily at 50 psi.

Investigation of the failure of aluminum gaskets showed three factors which might be held responsible: (1) poorly machined serrations in the flange faces which contact the gasket, (2) too large a gasket area, which prevented suffic-

FIG. 17
PARTIALLY ASSEMBLED REFLECTOR

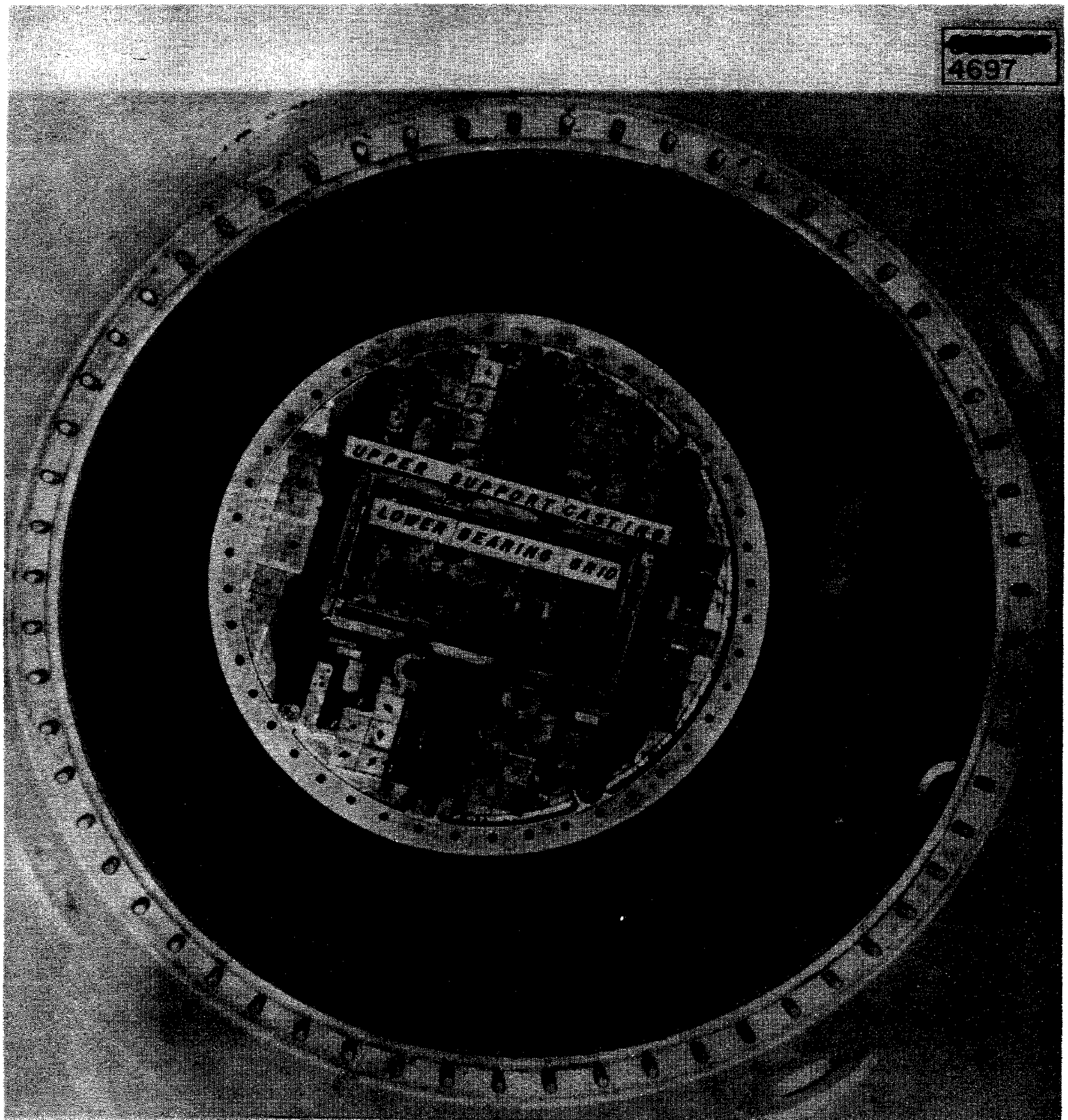


FIG. 18
COMPLETED REFLECTOR ASSEMBLY

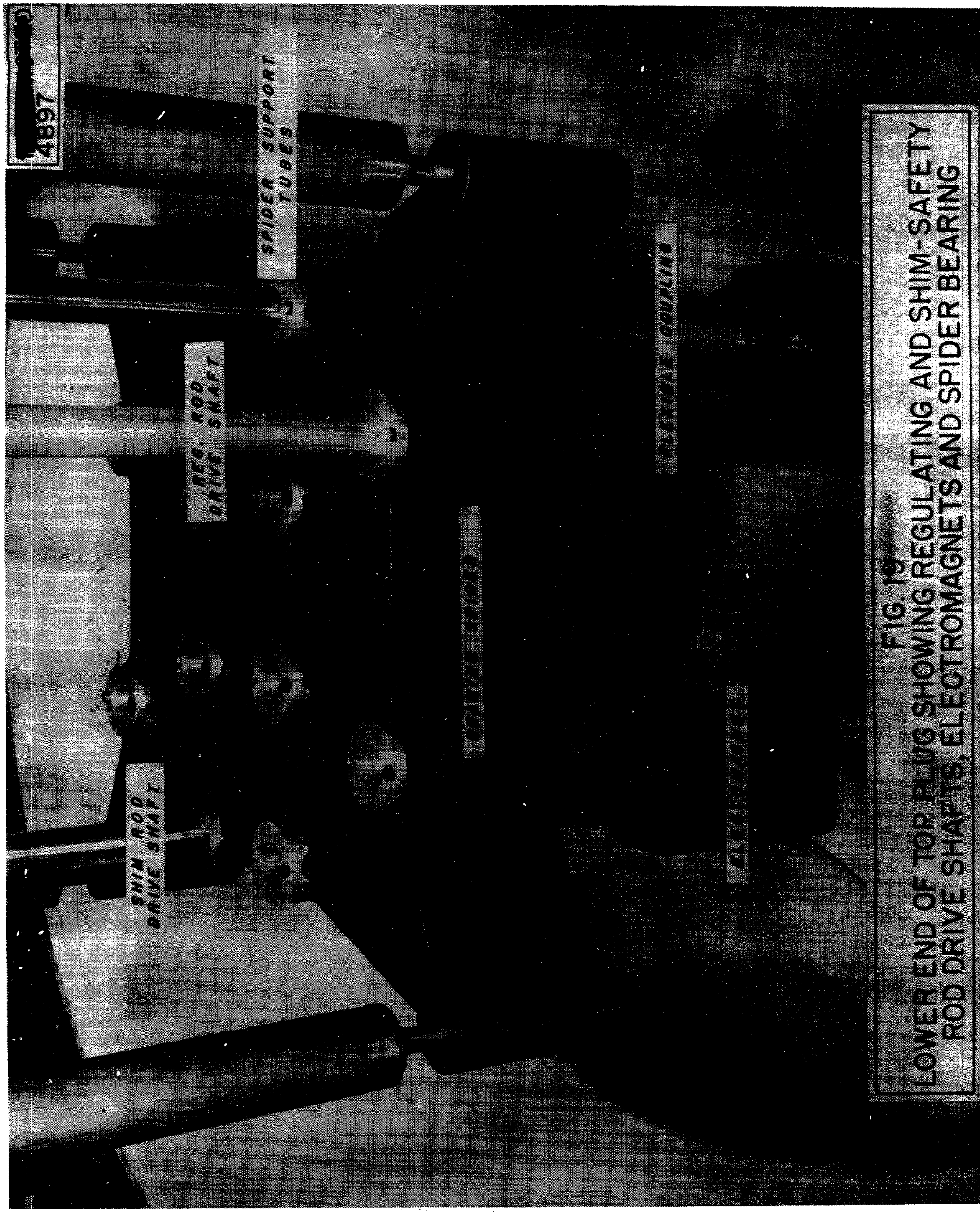


FIG. 19
LOWER END OF TOP PLUG SHOWING REGULATING AND SHIM-SAFETY ROD DRIVE SHAFTS, ELECTROMAGNETS AND SPIDER BEARING

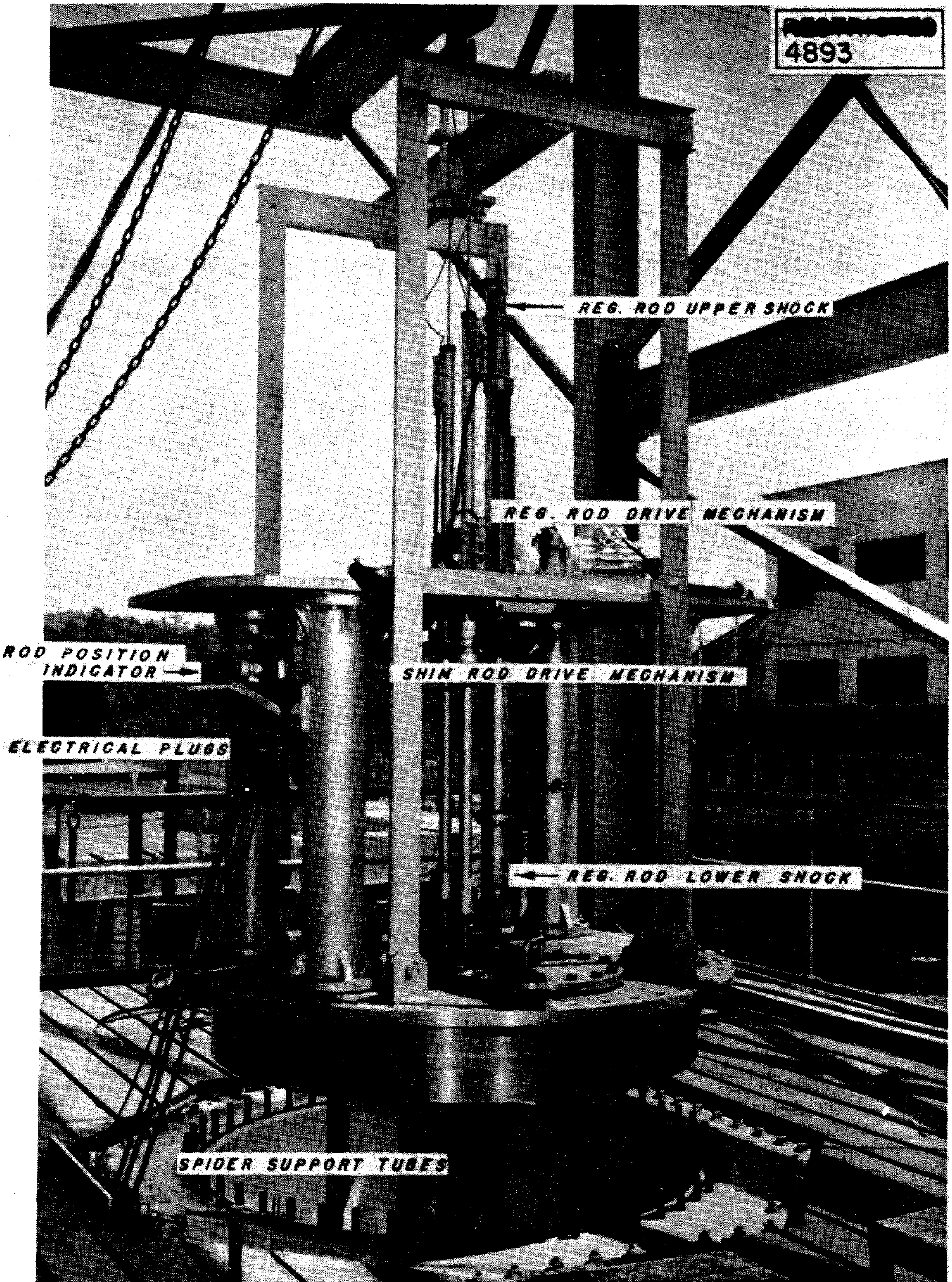


FIG. 20
TOP VIEW OF TOP PLUG SHOWING DRIVE AND INDICATING MECHANISMS OF REGULATING AND SHIM-SAFETY RODS

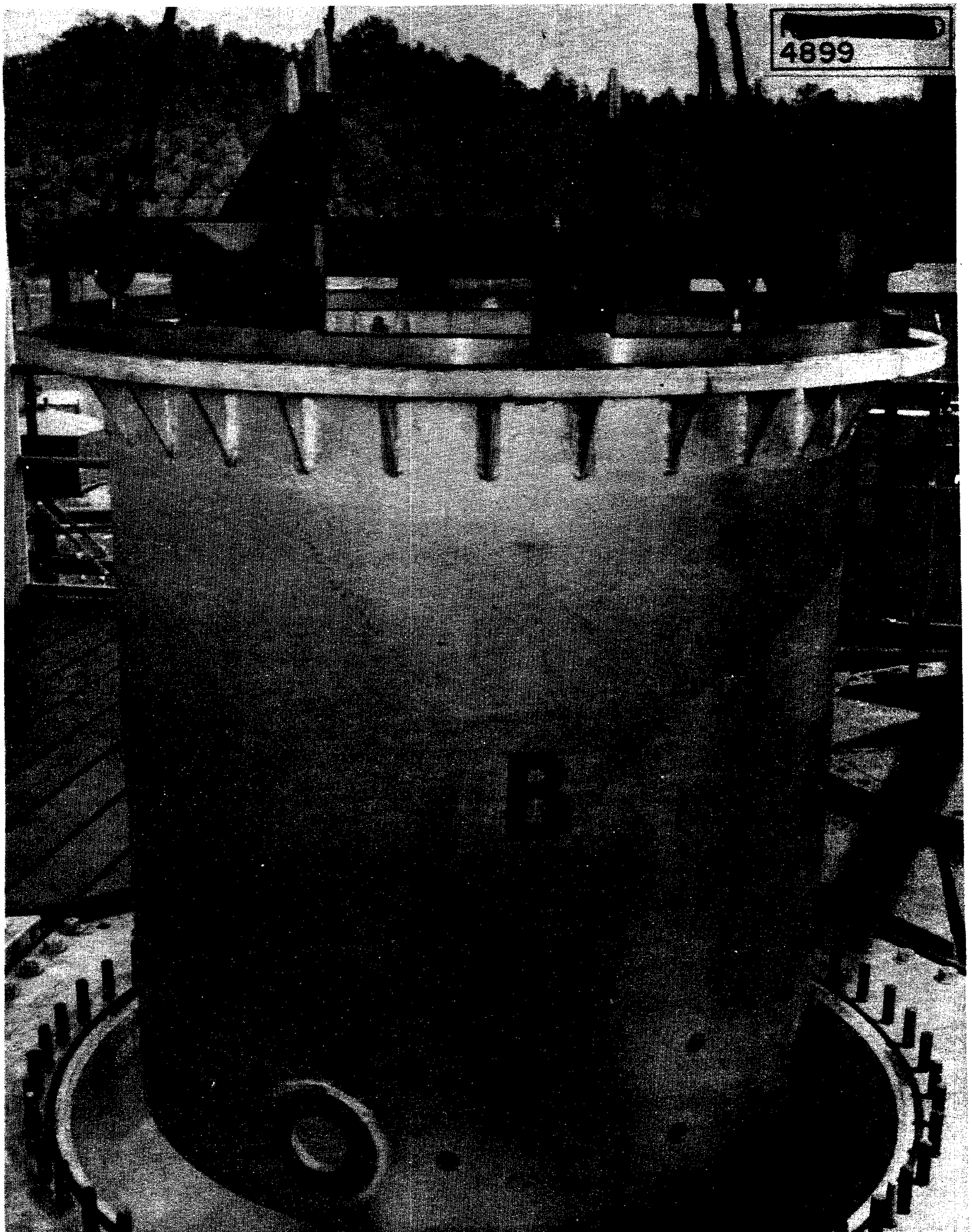


FIG. 21
SPIDER SUPPORT RING MOUNTED ON TANK SECTION B AND READY FOR LOWERING INTO SECTION A

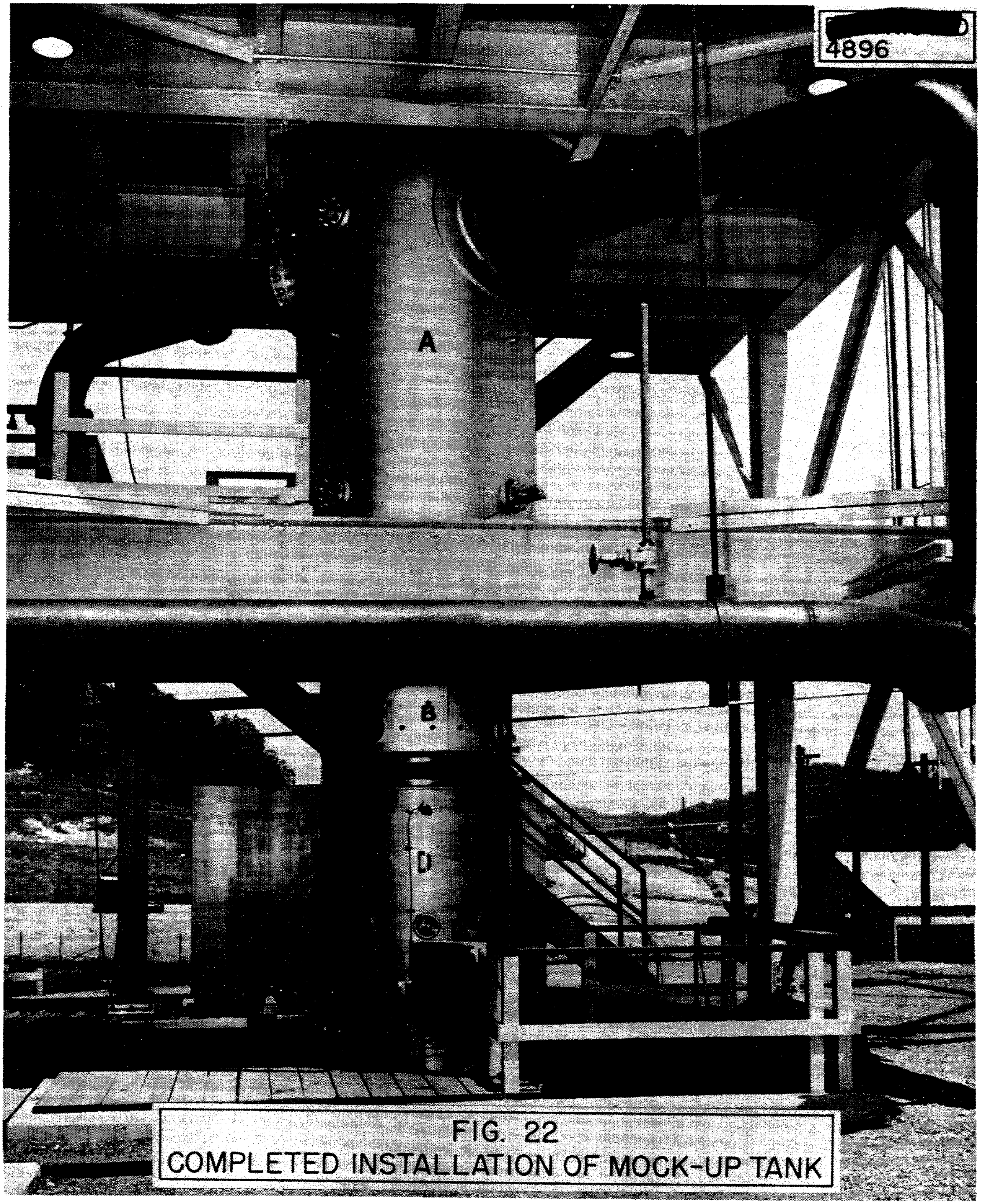


FIG. 22
COMPLETED INSTALLATION OF MOCK-UP TANK

[redacted]

ient force being applied per unit area of gasket with the bolt sizes designated, and (3) B and C sections flanges may not be heavy enough to prevent distortion with a single gasket.

Experimentation with 3/16 inch O.D. wire gaskets and 1/4 inch wide flat gaskets of annealed aluminum indicate that:

1. *A satisfactory seal can be had with the 1/4 inch wide flat gasket if spacers of the same thickness are employed on the opposite side of the bolt circle to prevent distortion. Minor defects in the serrated flange surface can be overcome in this particular design if the bolt loading is sufficient.*
2. *Wire gaskets should be satisfactory under the same conditions if the large diameter of the gasket can be held close enough (± 1/16 in.) so that serrations will not be crossed, However, the wire gasket is much more difficult to hold to such a tolerance than the flat gasket unless the flange is grooved for the wire, or serrations are eliminated entirely.*
3. *It is possible to stop leaks by pounding the outer edge of the aluminum gasket in the same manner that a lead joint is sealed.*
4. *Gaskets between well-machined flanges without serrations appear to seal better than between serrated flanges.*

Experimental work will continue until a completely satisfactory gasketing method is found.

MOCK-UP CONTROL

The problems of high flux reactor control have been discussed in many reports. The article "Shim Rod Control" by E. P. Epler, J. D. Trimmer, and H. W. Newson in CNL-35, describes the model from which the mock-up control circuits were derived.

The primary purpose of the mock-up is to test the mechanical and electrical design of the reactor control components. Many of the safety circuits and all flux-level and reactor-period instruments have been omitted. However the complete reactor control problem has been kept in mind throughout, and should the additional instrument and safety circuits be required they may be installed at any time.

To facilitate operation a console type control board has been used for the mock-up (Fig. 2). The operator has all controls within easy reach and can operate the equipment continuously without excessive fatigue. When the additional controls are added this board may be used as a training board for reactor operators.

[redacted]

[redacted]

The present mock-up control circuit as shown in Fig. 23 permits the withdrawal and insertion of the shim rods, permits position control of the regulating rod, and provides for continuous testing of all rods. In continuous test the shim rods are dropped, the drive mechanisms run down and engage the rods, the rods are withdrawn to their upper position, dropped, etc. In this manner the rods and associated equipment receive the equivalent of months of operation in a relatively short time, without continual operator attention. The circuit is arranged so that both rods may be tested simultaneously or either rod may be tested without interfering with the manual operation of the other rod. Continuous testing of the regulating rod drives the rod up and down between arbitrarily set upper and lower limits by means of the motor driven positioning rheostat.

MOCK-UP HANDLING TOOLS

Several special remote type handling tools and accessories have been made and used in assembling and disassembling the active section of the reactor mock-up.

The use of springs between the hoisting mechanism and the handling tools is being studied to obtain finer control over the load, especially where close fits between mating parts are required.

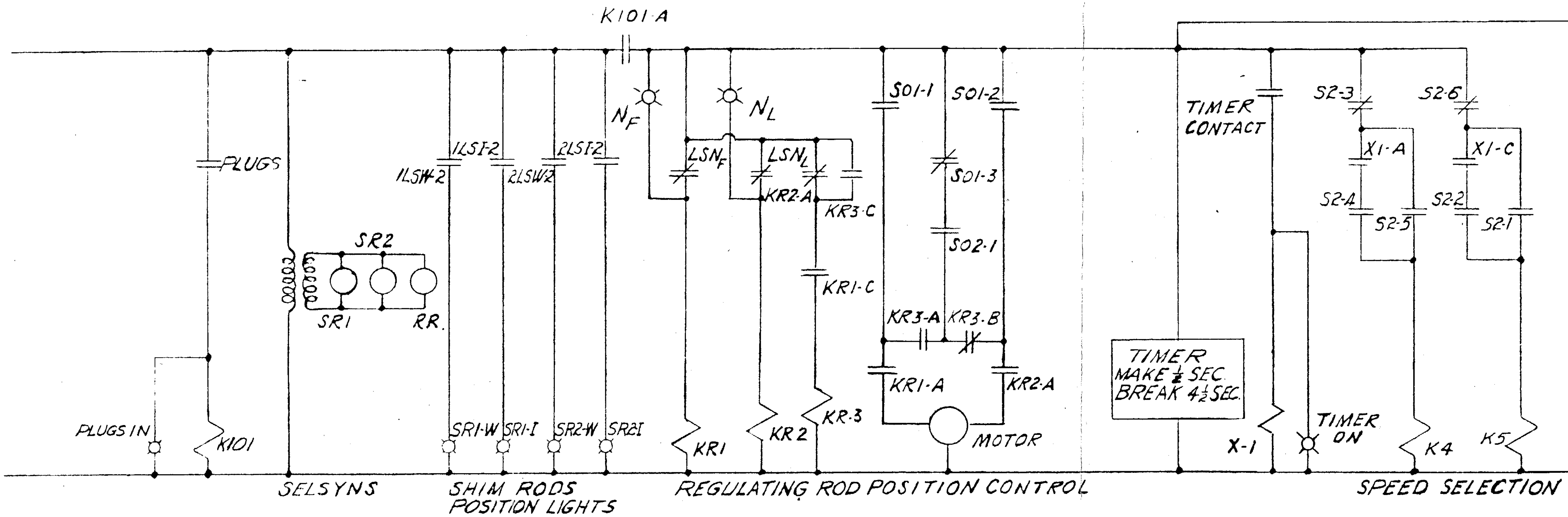
The design of wrenches for remote bolting and unbolting of flanges inside the pile tank assembly is in a preliminary stage.

The tools are illustrated in Figs. 24-29. Tools are approximately 20 ft long. The method of coupling to the hoisting device is a conventional eye. Handles on the tool shaft at the hoisting mechanism are conventional.

Operating Platform. (Fig. 24). This accessory is designed to provide a safe method of working directly above any part of the open tank A during removal of reactor parts. The platform is supported on casters while being moved. When the platform is in position over the reactor mock-up, the casters are raised by screws and the platform rests on rigid legs. An 18 inch opening permits removal of all but the larger pieces but partly guards an operator against falling into the tank. The present platform is heavy and unwieldy, and will be modified.

Regulating Rod Tool. (Fig. 25). This aluminum tool is approximately 22 ft long. The tapered end of the regulating rod shown is the end which extends above the upper support casting after the driving portion of the regulating rod has been unscrewed and removed with the top plug assembly.

[redacted]

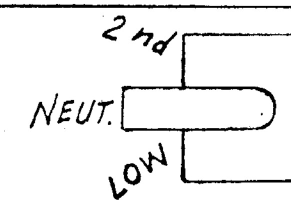


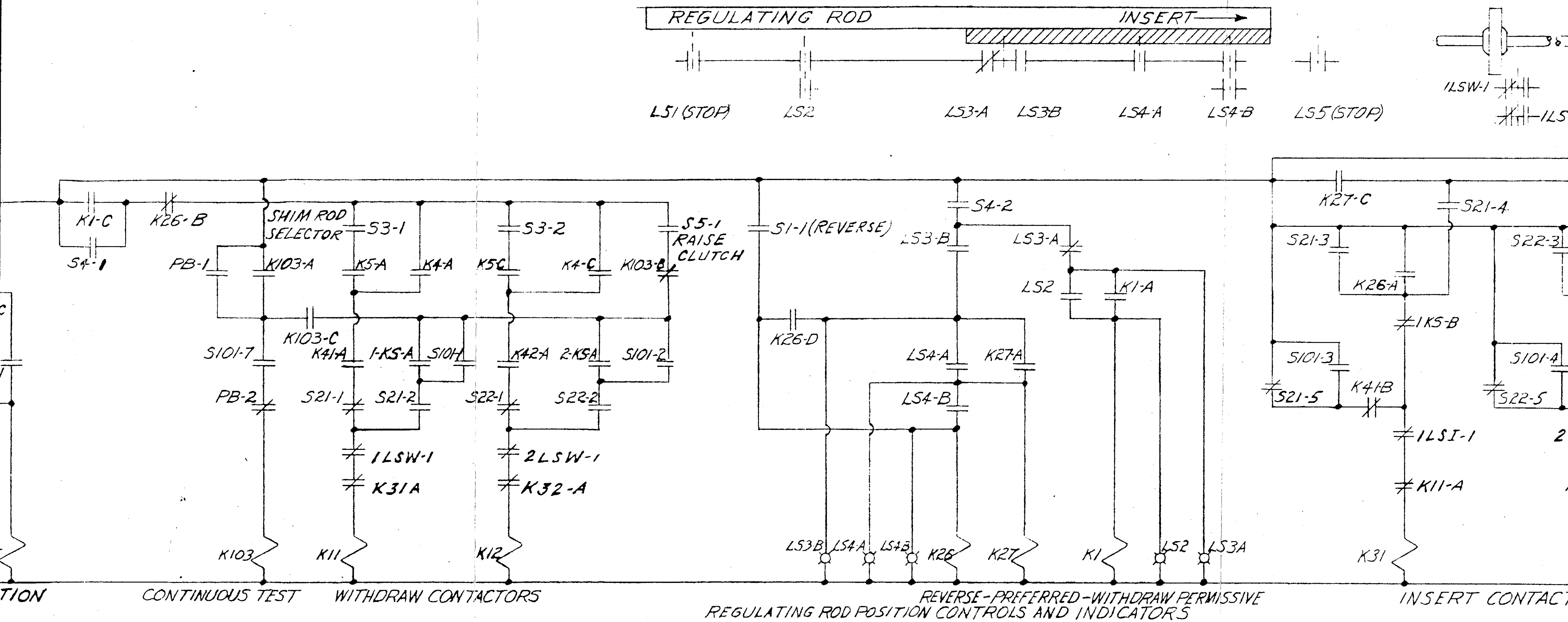
S01 - REGULATING ROD POSITIONING

CONTACT	POSITIONS SR RAISE	NR OFF	SR LOWER
1	X		
2			X
3		X	

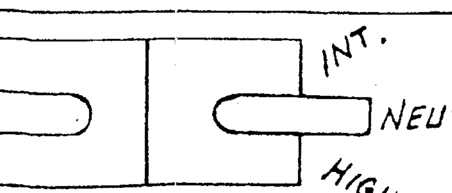
S-2 - TWO S

CONTACT	POSITIONS SR LOW	NR NEUT.	SR 2nd.
1			X
2	X		
3		X	





S-2 - SPEED SHIFTER
Two Switches back to back.



CONTACT	POSITIONS NR INT.	NR NEUT.	NR HIGH
4	X		
5			X
6		X	

S4 - SERVO-MANUAL CONTROL

CONTACT	POSITIONS NR ROD 1	NR MANUAL	NR ROD 2
1	X		
2		X	

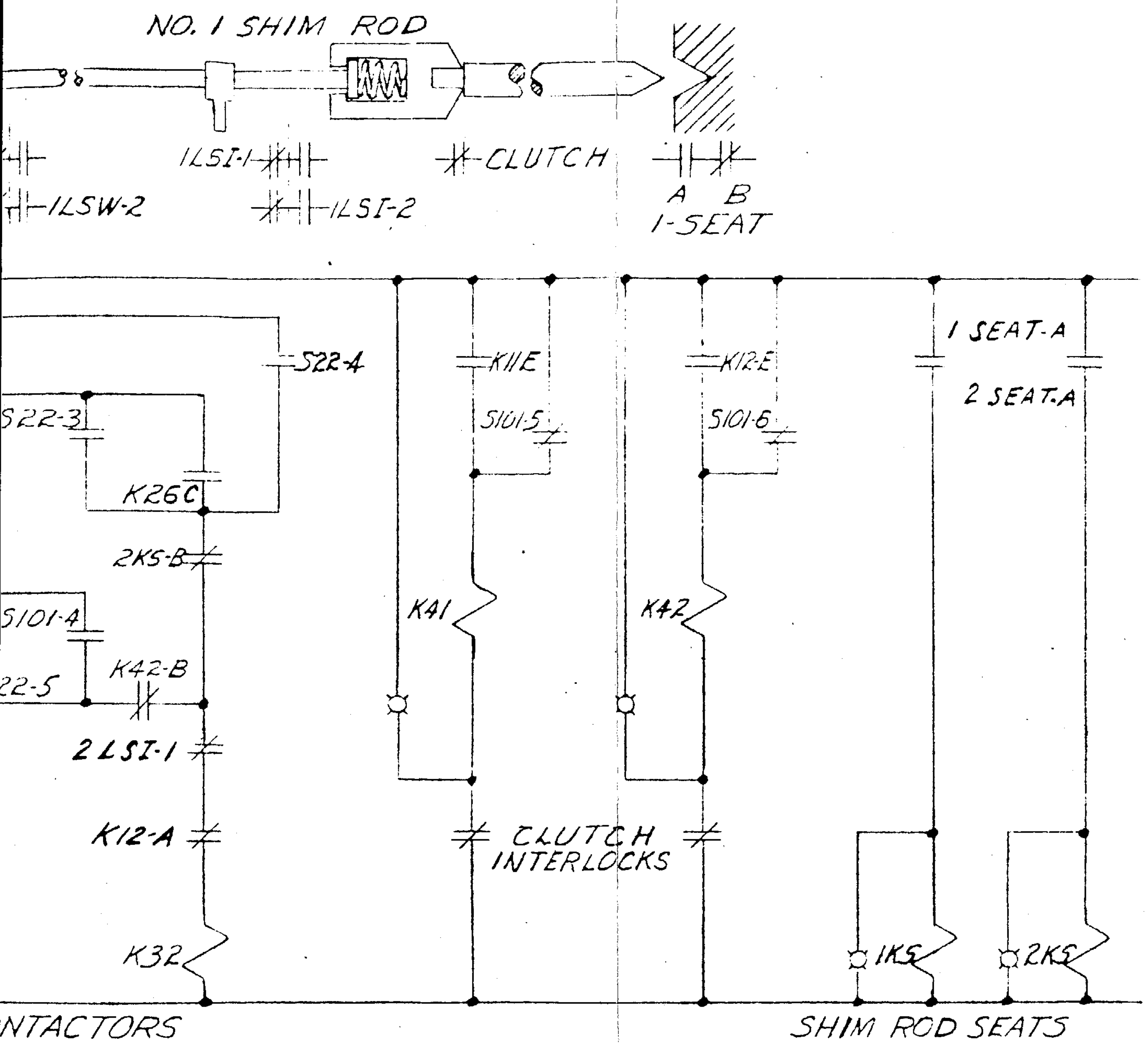
S21 & S22 - SHIM ROD CONTROL

CONTACT	POSITIONS NR RAISE	NR OFF		NR BLOCK	NR NORM.	NR PREF.	SR JOG
1					X	X	
2	X						
3							X
4						X	
5				X	X	X	X

S101 - SHIM ROD CONTINUOUS TEST

CONTACT	POSITIONS NR ROD 1	NR OFF	NR ROD 2	NR 1 & 2
1	X			X
2			X	X
3	X			X
4			X	X
5		X	X	
6	X	X		
7	X		X	X

DATE	CO



NOT CLASSIFIED

REVISIONS

DATE	CODE	CHANGE	CHK.

LIMITS ON DIMENSIONS UNLESS OTHERWISE SPECIFIED
FRACTIONS ± ~
DECIMALS ± ~
ANGLES ± ~

FIRST USED
DESIGNED J.E. OWENS 4-5-49
DRAWN BY OWENS, HURAY.
STARTED 5-10-49
FINISHED 5-11-49
CHECKED 5/11/49
APPROVED

OAK RIDGE NATIONAL LABORATORY
INSTRUMENT DEPARTMENT

FIG. 23
MOCKUP CONTROL
ELEMENTARY DIAGRAM

SCALE NONE

Q-882-1

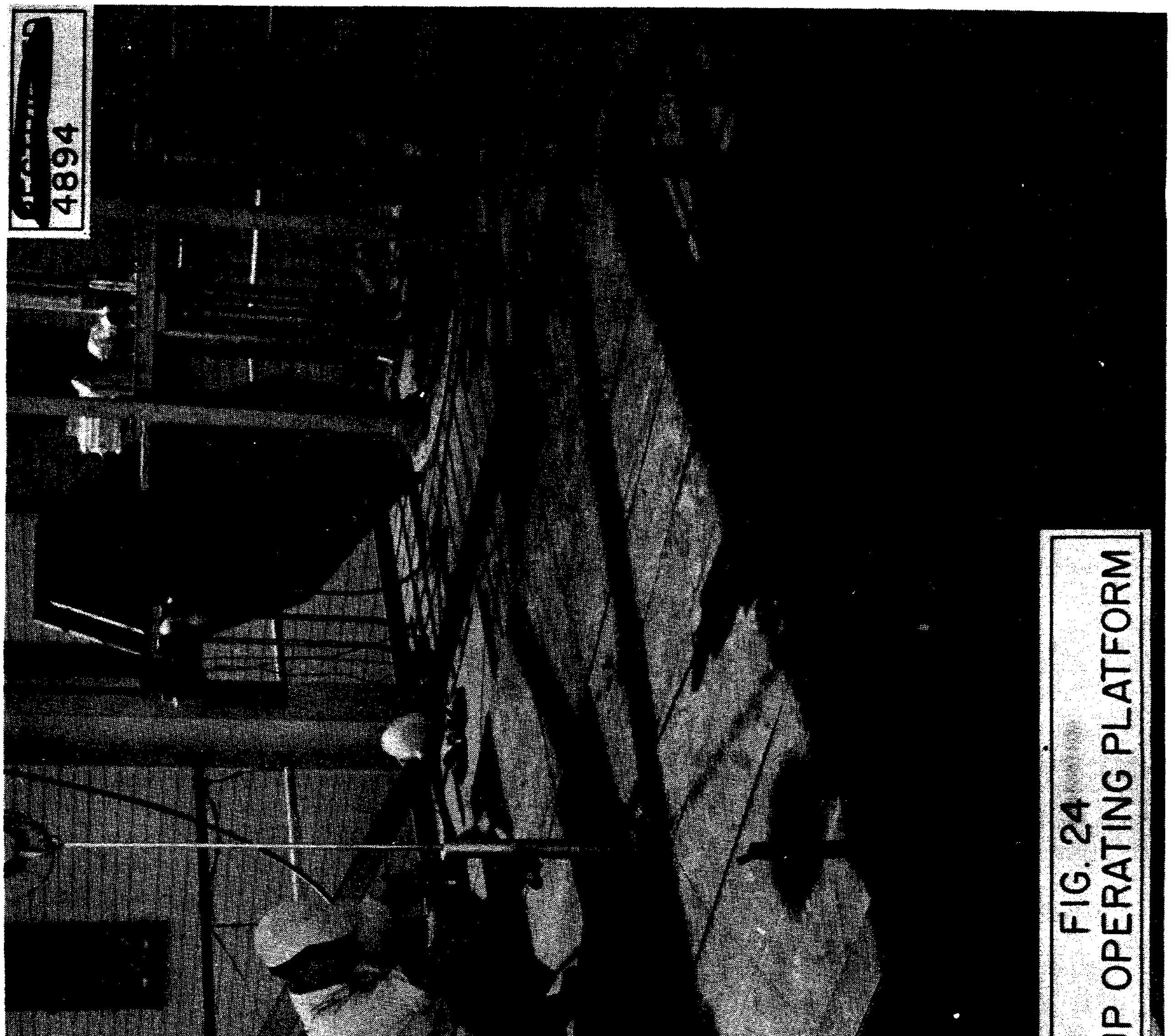


FIG. 24
MOCK-UP OPERATING PLATFORM

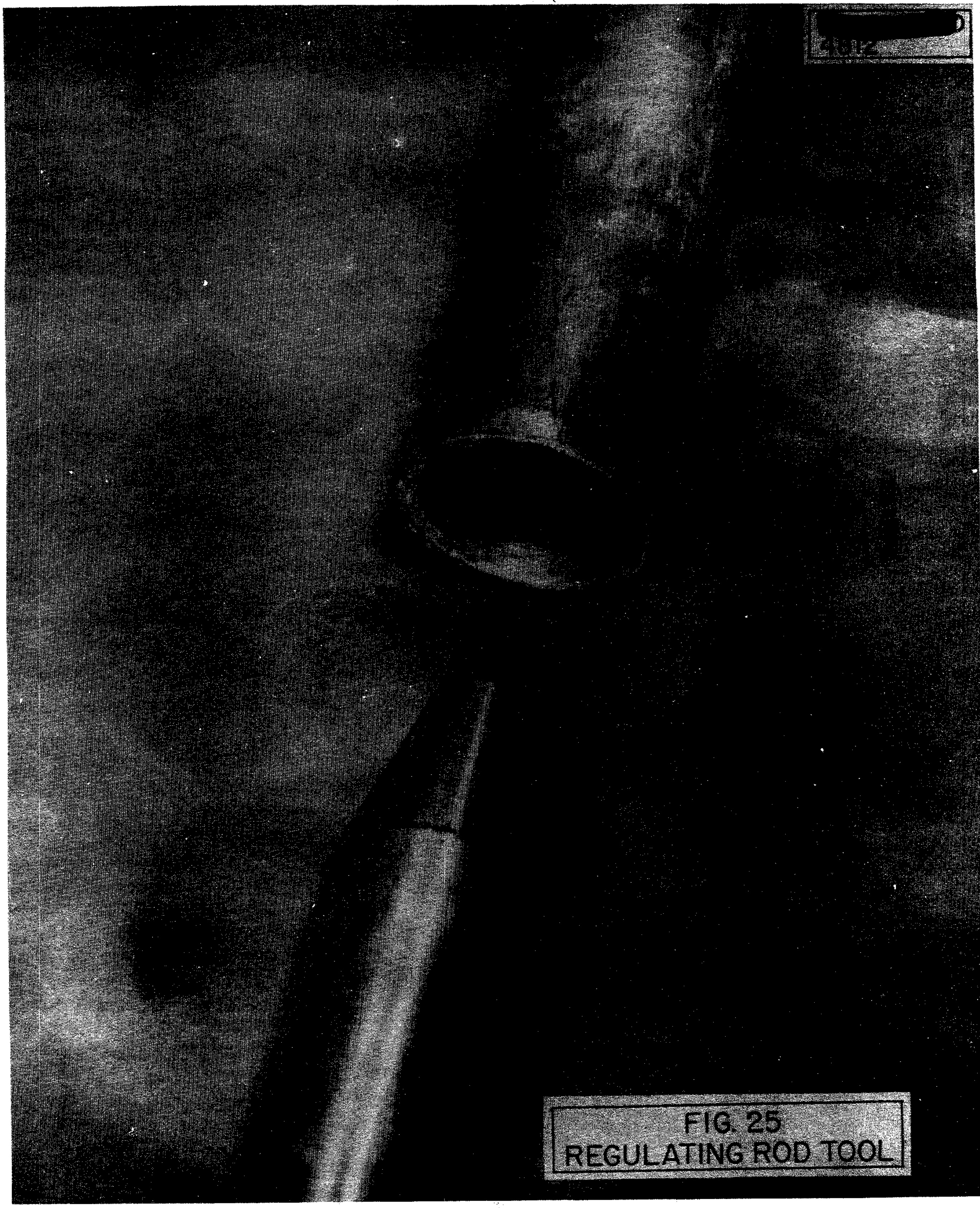


FIG. 25
REGULATING ROD TOOL

The tool has a cylindrical sleeve which engages the regulating rod and centers it with relation to the tool. A stainless steel stud bolt with a smooth radius at the leading end is located in the center of the tool and provides a means of coupling to the regulating rod threads without damage to its threads.

Shim Safety Rod Tool. (Fig. 26). This tool was fabricated to remove the portion of the shim safety rod beneath the magnetic coupling. The illustration shows the tool inserted into the water port just beneath the armature insert. The milled 45° slot in the tool engages the conical 45° section of insert, and firmly positions the tool. The part of the tool which extends above the water port in the shim safety rod is an extra precaution to prevent the tool from disengaging the rod.

Tool to Handle the Upper Grid Assembly. (Fig. 27). A special tool which is used to unlock and lift, as well as to lower and lock the assembly, is shown coupled to the lifting bar of the locking mechanism. The tool is essentially a double hook, designed for a minimum of side slippage on the locking bar, and so that coupling can be made at approximately 18 ft under water. The long guides which extend from the lower hook section, facilitate proper mating with the locking bar.

Active Assembly Lifting Tool. (Fig. 28). In this illustration only the coupling and end of the tool are shown. Before entry is made into an assembly the two pins shown are moved to a position slightly beneath the pilot housing. This is accomplished at a point approximately 22 ft away from the pilot by means of a trigger connected to a cable, which operates levers moving the pins. A spring extends the pins when they have made entry into the two holes shown in the assembly.

The pilot is provided with vents to permit cooling water to flow through the assembly piece after coupling is made.

Reflector and Discharge Plug Lifting Tool. (Fig. 29). The picture illustrates a simple tool in the form of a hook, which is used to engage the eyebolts that have been provided in all A type reflector pieces and in the discharge plug. Also shown is the rack designed to support the discharge chute plug within the B tank when the discharge chute is in use. The flange on the upper part of the rack is bolted to the flange on the spider support casting.

In the design of the rack the clearance between the plug and rack was held to approximately 1/8 inch on each side. A pin is provided in the bottom of the discharge plug to mate with a hole in the bottom of the rack.

The rack was positioned in tank B at a point such that no interference with the removal of A type reflector pieces would occur.

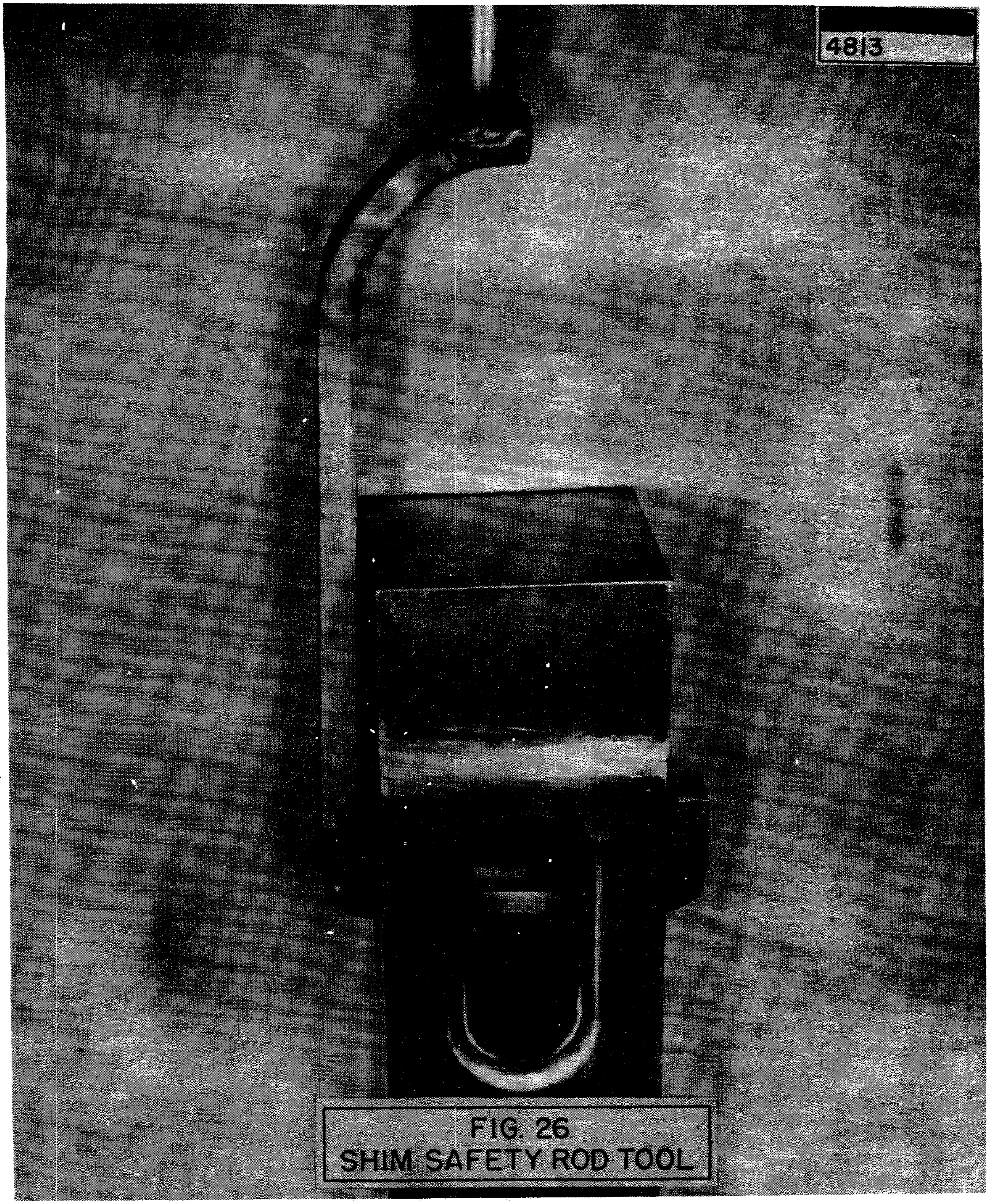


FIG. 26
SHIM SAFETY ROD TOOL

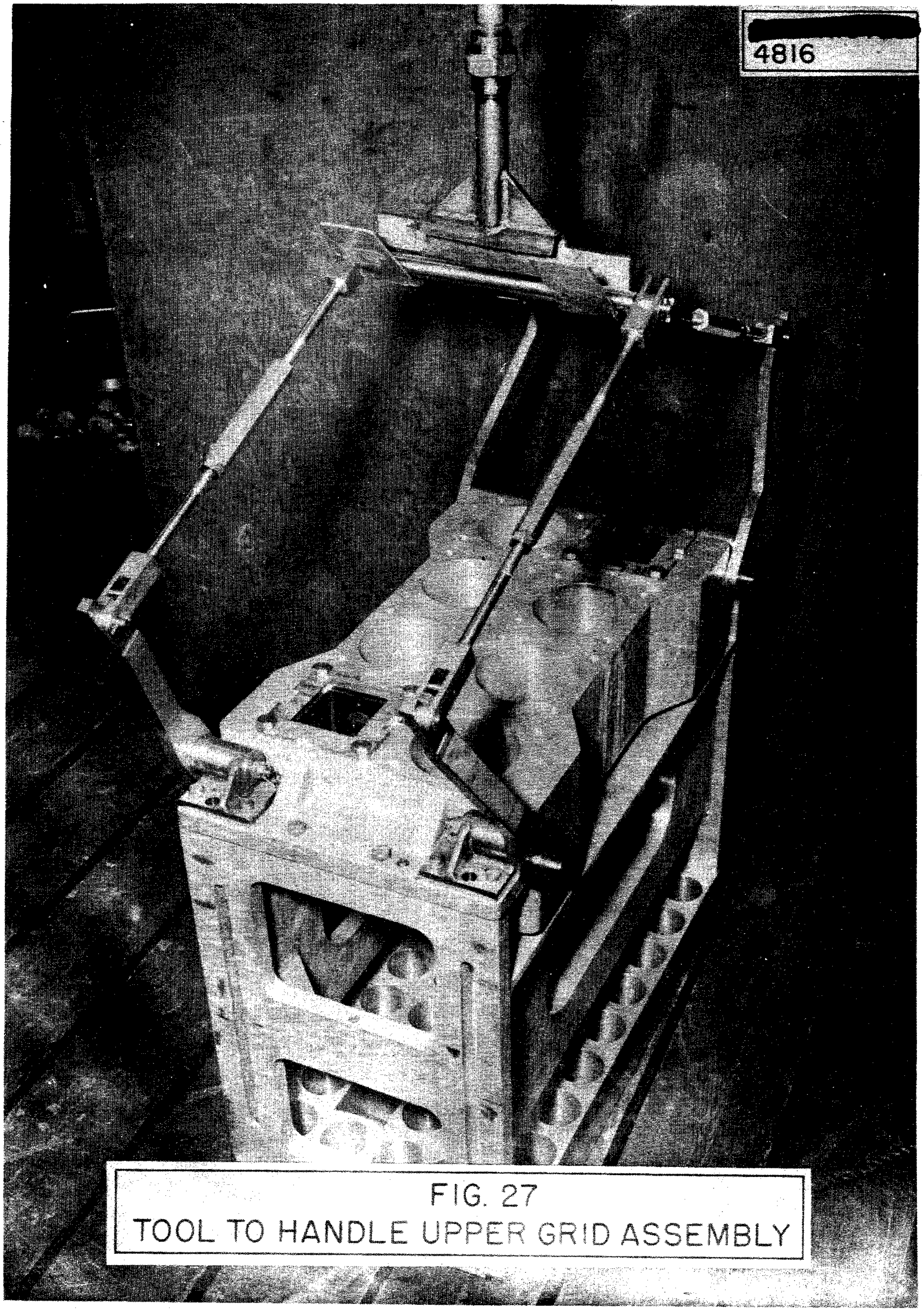


FIG. 27
TOOL TO HANDLE UPPER GRID ASSEMBLY

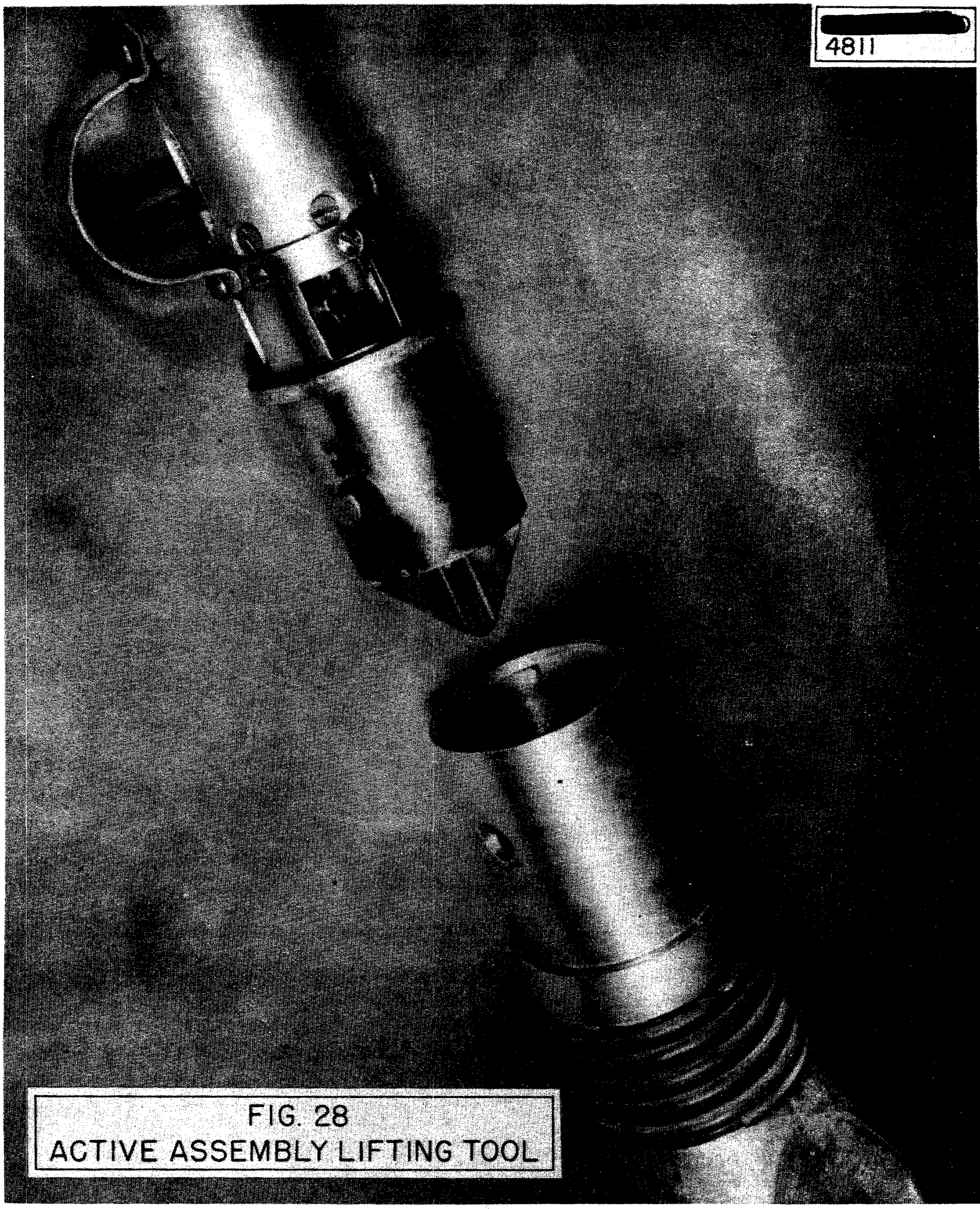


FIG. 28
ACTIVE ASSEMBLY LIFTING TOOL

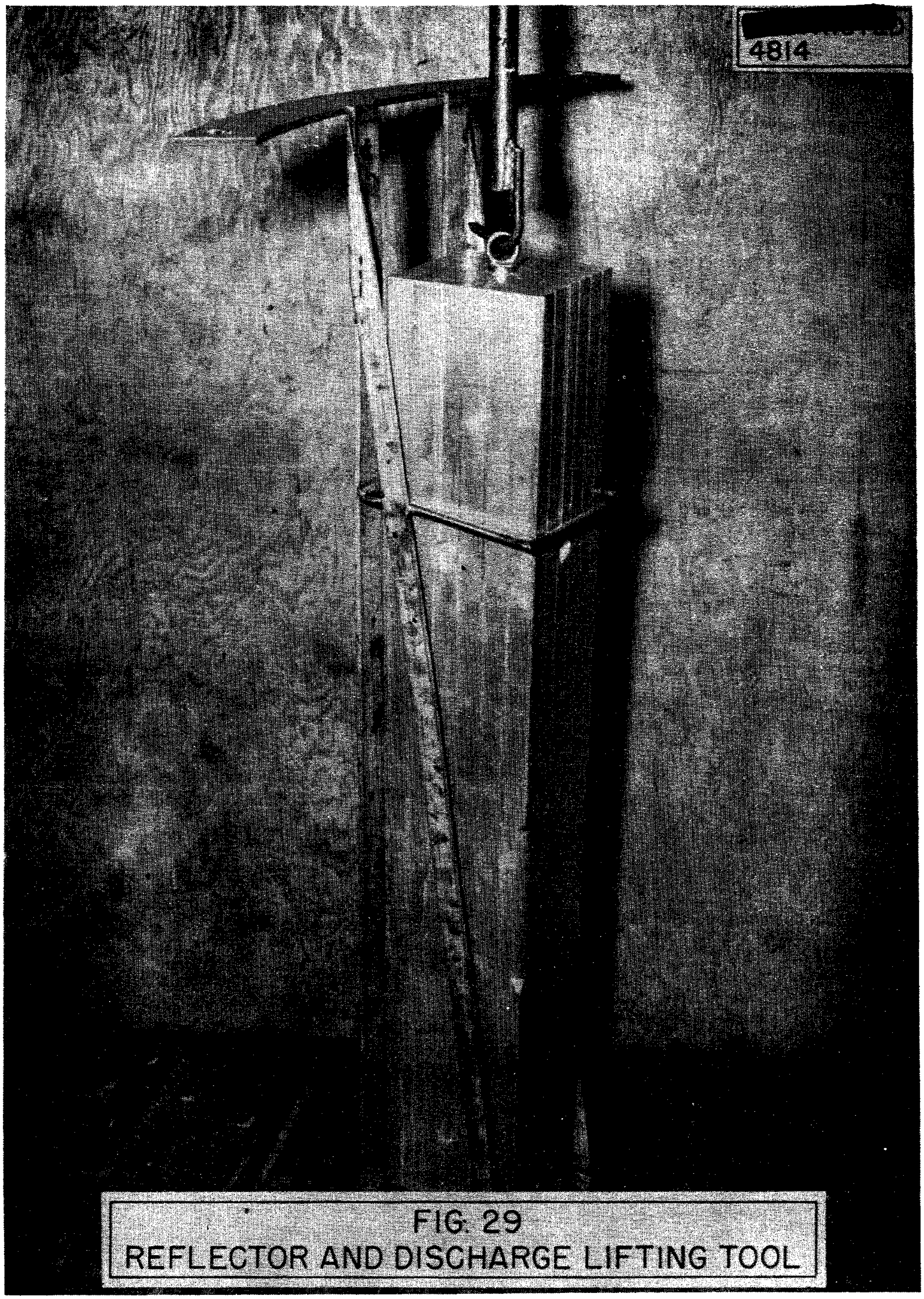


FIG. 29
REFLECTOR AND DISCHARGE LIFTING TOOL

STRAIN MEASUREMENTS

Baldwin-Southwark SR-4 strain gages have been attached to the following component parts of the reactor mock-up for strain measurements during operation: tank section D, lower cradle, upper support casting, lower grid support, upper assembly grid, lower assembly grid, and side skirt plate.

From these measurements, maximum, minimum, and shear stresses will be calculated.

DISTORTION MEASUREMENTS

Reference measurements have been made on the following parts of the mock-up to determine any dimensional changes due to creep or overstress during a long term operating period: lower grid support, upper support casting, lower assembly grid, upper assembly grid, side skirt plates, and tank section D, inside diameter.

Changes in these reference dimensions will be noted on disassembly of the mock-up.

Changes in the diameter of tank section D under operating conditions will be determined by outside diameter measurements at a number of points, at regular time intervals. A metal band, free to move at one end, has been wrapped around the tank to check circumferential expansion.

HYDRAULIC TESTS

Preparation for the mock-up hydraulic tests is nearing completion. After having been completely assembled for demonstration, the reflector has been removed from the mock-up for installation of the flow measuring devices described in ORNL 323, Part I. All machine work required for installation of these devices has been completed, and all tubes for differential pressure measurement have been installed in the appropriate reflector pieces. All plastic mountings for the thermistors have been made, and installation of the thermistor cells is in progress. Indicating instruments for the fluid flow measurements are being installed on a panel in the mock-up control room.

A report of experimental work at Oak Ridge National Laboratory by an MIT Practice School group on the use of thermistors for velocity measurement has been issued as MIT Practice School Memorandum EPS-X34, March 30 1949, entitled 'A Thermistor Study.' The report gives an empirical correlation, with a maximum error of 35% in estimation of velocity. Although the correlation is not as good as desired, it will be useful for flow measurements in the mock-up.

GALLING OF STAINLESS STEEL

Previous testing of stainless steel nuts and bolts indicated the possibility of galling at points where two stainless steel surfaces of equal hardness are in moving contact. On the basis of this information, all nuts for use in the mock-up were specified as the hardened type 416 stainless steel. Through error unhardened type 416 nuts were obtained and their hardness was not checked until after use in the mock-up. Experience to date with the use of these nuts as well as the type 304 unhardenable variety has neither proved nor disproved the apprehension that galling might be troubleseome. The only nuts that have galled were probably overloaded due to misalignment. However, it is felt that the danger should be minimized by substituting the hardened stainless steel for one of the two mating surfaces where the possibility of galling exists. The necessary parts are being procured.

Galling has also been encountered at several other points where mating 304 stainless surfaces have been heavily loaded. In each case it is probable that the surfaces were overloaded due to misalignment or some other extenuating circumstance, but it is felt that galling would not have occurred if the surfaces had been dissimilar in hardness. Further experience should result in clarification of this subject.

CORROSION OF REACTOR MATERIALS

BERYLLIUM

Accelerated Corrosion Tests. Efforts were made to develop an accelerated test to predict, at least qualitatively, the susceptibility of beryllium to corrosion particularly of the type which produces surface blisters. Of the methods tried, only the high temperature autoclave test showed promise.

In the unsuccessful attempts, a sample was taken from a piece of extruded beryllium of decidedly inferior quality, which was known to have formed blisters previously in both flow corrosion tests and on six months exposure to normal indoor atmosphere. A machined specimen, about 5/8 inch diameter, was tested as follows with the results indicated:

1. *150 hours exposure to non-aerated water vapor at 100° C; weight loss, 0.9 mg.*
2. *102 hours exposure to aerated water vapor at 100° C; weight gain, 1.9 mg.*

[redacted]

3. *24 hours exposure to water-saturated air passing over specimen at 300° C; weight loss, 3.4 mg.*

4. *12 hours intermittent exposure to a steam and air mixture passing over the specimen at 300° C; weight gain, 0.4 mg.*

No blisters were produced in any of these tests. This sample was then re-machined and used as specimen 7 in the autoclave tests.

Autoclave Tests. Eleven samples of various types of beryllium were subjected to a temperature of 269° C and a pressure of 780 psi absolute. The time of test, including preheating and cooling, was 144 hours. Demineralized water was used for the test medium. The initial water *p*H was 5.5; the final *p*H was 9.9. The marked increase in *p*H was attributed to the fact that the glass hooks used to hold the test specimens were disintegrated under autoclave conditions. With two exceptions, the general condition of the test specimens removed from the autoclave was good, although there were white deposits on the surfaces. Since there were no pits beneath the deposits, it is possible that the white products originated from the decomposition of the glass hooks.

Specimen 7, a material known to be susceptible to blistering, even during air exposure, showed several surface blisters as well as numerous fine cracks parallel to the extrusion direction. Sample 3 was taken from the same material used in the long term corrosion tests described later. After exposure, it showed a crack on one flat surface about 1/8 inch from the outer circumference. This crack extended for 2.5 cm parallel to the outer edge.

Attempts to clean the white deposits from the specimens met with negligible results. It was impossible to clean thoroughly without removing the metal also, so quantitative measurements of corrosion damage were not feasible.

The specimens were re-polished to remove adhering products, then returned to the autoclave for continued testing, this time suspended in the test chamber by fine tantalum wire. These results are not yet available.

The various types of beryllium used in this test are described in Table 1.

Long Term Tests. Figure 30 shows the continued growth of blisters on extruded beryllium exposed for 668 days at 85° C to demineralized water containing 0.005 *M* hydrogen peroxide. This test, XB-5, was previously summarized in ORNL 323, Part I. The test started May 21, 1947, on an assembly of five 2-1/4 inch diameter extruded beryllium discs bolted together. The beryllium, obtained in 1947, had an extrusion reduction ratio of approximately 8 to 1. Metallographic examination shows the material to be quite impure and not representative of good extruded material.

[redacted]

[redacted]

TABLE 1

Beryllium used in autoclave test

SAMPLE NUMBER	DESCRIPTION OF BERYLLIUM
1	Q process metal received from Brush Beryllium Co. early in 1947; samples received in a machined condition, 1/4 in. x 1 in. x 2 in.; density, 1.802; no chemical analysis available.
2	Extruded beryllium received from MIT during latter part of 1946; received as a steel-jacketed bar, 1-1/8 in. diameter and 5 ft long; extrusion ratio, approximately 16:1; density, 1.855; material used for major portion of beryllium corrosion studies during 1946-1948; no chemical analysis available.
3	Extruded beryllium received in 1947; sample received as a 3 in. diameter bar; extrusion ratio approximately 8:1; samples machined to 2-1/4 in. diameter by 1/4 in. thick; showed poor machining qualities such as edge-tearing; density, 1.851; no chemical analysis available; metallographic examination shows this to be high in inclusions and well below average quality of MIT extruded beryllium.
4	Extruded beryllium received from MIT in 1947; history unknown; samples received in machined condition; size, 1/4 in. x 1-1/4 in. x 2-1/4 in.; density, 1.85; no chemical analysis available.
5	QM beryllium received from Brush Beryllium Co. in 1949; Bar No. 4382BPQM; sample size 3/16 in. x 1-1/16 in. x 2 in.; machined from 1 in. x 4 in. x 6 in. block; density, 1.849.
6	Extruded beryllium bar No. 788; extrusion ratio approximately 6.5:1; sulfuric acid etch disclosed cracks in billet; specimen cut 6 in. back of A end.
7	Extruded beryllium obtained in 1947; machined from 1-1/8 in. diameter bar stock; history unknown; machined sample blistered during six months exposure to air; sample re-machined for autoclave test; density, 1.86.
8	QM beryllium received from Brush Beryllium Co. Bar No. 6, Lot 3.
9	QM beryllium received from Brush Beryllium Co. Bar No. 7, Lot 4.
10	Beryllium bolt material received from MIT; extrusion ratio approximately 16:1; density, 1.858; annealed condition.
11	QM beryllium received from Brush Beryllium Co.; magnesium content, 1750 ppm.; Lot No. 2.

All samples were belt surfaced on No. 80 emery paper and then defilmed in cold, 25 percent HNO_3 for 20 minutes prior to testing. The densities of the metals were determined by successive weighings in air and water.

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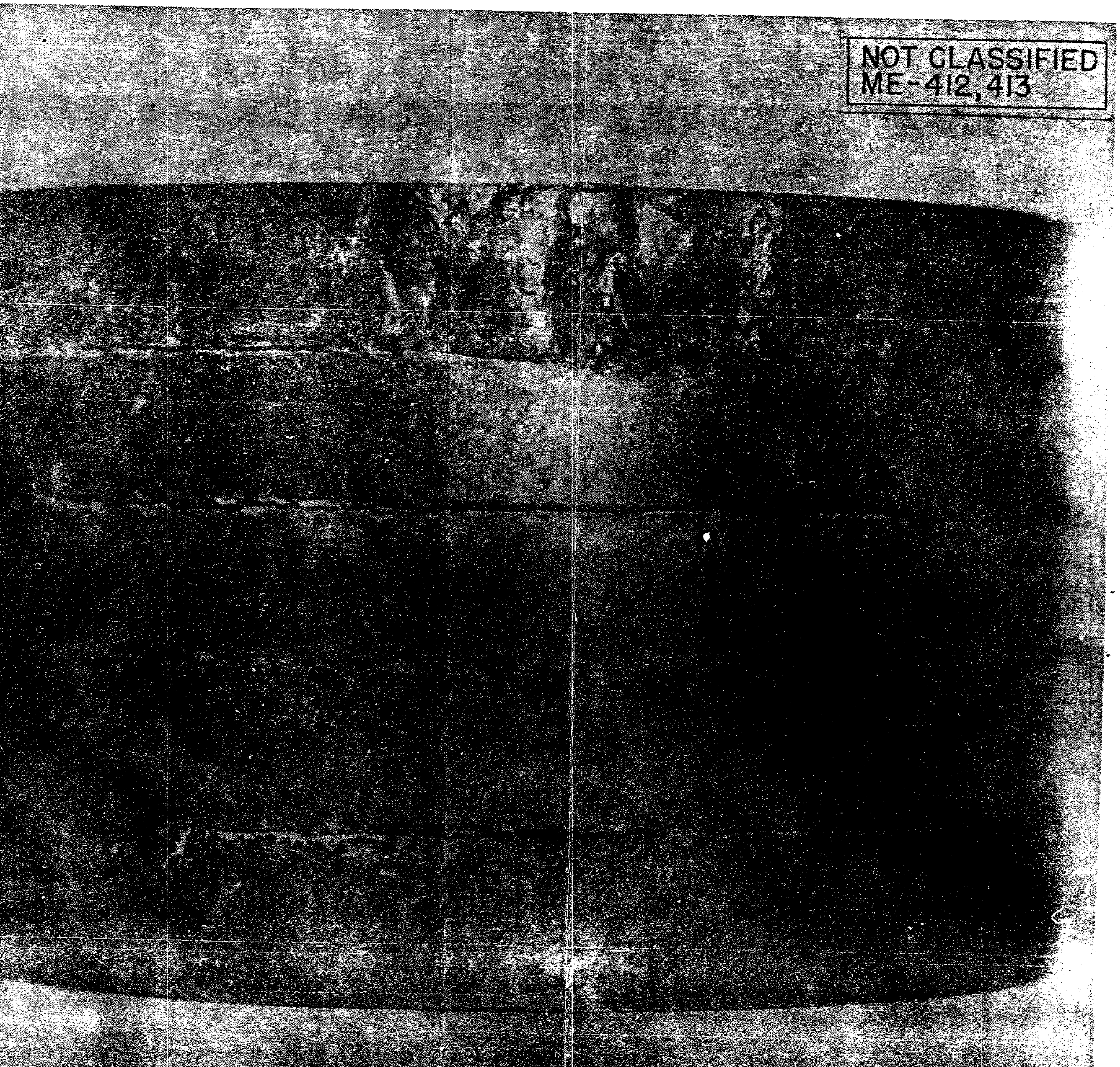
BLISTER FORMATION ON EXTR



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ED BERYLLIUM AFTER [illegible] DAYS

The following tabulation compares data collected after 569 days exposure with data obtained after 668 days.

	569 Days	668 Days
Number of blisters	*3*	*9*
Length of largest blister	*240 mils*	*640 mils*
Number of cracks	*1*	*1*
Length of crack	*280 mils*	*290 mils*

The average expansion at different intervals due to corrosion products between the interfaces is shown below:

Exposure, Days	Average Interfacial Expansion mils/interface
45	*0.52*
104	*0.88*
162	*0.88*
193	*1.00*
344	*1.52*
480	*3.56*
569	*3.60*
668	*4.32*

Numerous blisters ranging from 60 to 160 mils in diameter formed on the circumferential surfaces of the beryllium discs during the last 99 days of exposure. This was an increase of 6 blisters over the last reported data. The largest blister extended 120 mils above the surrounding metal surface. There was an increase in length of 10 mils in the crack (Fig. 30) during the last 99 days. This test will be dismantled and examined, and a report will be issued at a later date.

Corrosion of a Replaceable Beryllium Assembly. A test to observe the susceptibility of a replaceable beryllium assembly to corrosion when exposed to simulated reactor cooling water, moving at a velocity of 15 ft/sec, has progressed for 240 days. No pitting or interfacial corrosion deposits have been observed. The cooling channels have remained bright and free from corrosion products. The assembly has exhibited no significant general corrosion during this exposure. The assembly was made from three 1 in. × 3 in. MIT extruded beryllium pieces, machined and fastened together with beryllium bolts.

[redacted]

QM Beryllium Corrosion Test Program. Various phases of the proposed corrosion testing program for QM beryllium metal have been started. The tests now in progress include:

1. *Long-time exposure tests on QM metal (Lot 1, Y-4382BPQM) containing 1.86 percent BeO. Tests will be continued for one year and results will be compared with 16:1 reduction ratio extruded beryllium exposed in the same solutions. Specimens will be removed monthly. Test media are peroxide-free demineralized water, and demineralized water containing 0.005 M hydrogen peroxide, both at 85° C.*
2. *Sixty day tests with QM metal (Lot 2) containing 1750 ppm of magnesium and 1.21 percent beryllium oxide. Single and couple tests with 2S and 356 aluminum alloys in demineralized water containing 0.0005 M and 0.005 M hydrogen peroxide, at 85° C, are in progress.*
3. *Sixty day tests with QM metal produced by the envelope process, using 300 psi pressure (Lots 3 and 4). Tests are similar to those described in (2).*
4. *A dynamic corrosion test on a full-sized QM beryllium assembly will be started as soon as material is obtained from the Brush Beryllium Company.*

CORROSION OF FUEL ROD ASSEMBLIES

Four transverse sections, one inch long, were cut from a full-size active fuel rod assembly. These specimens were exposed to various test media to determine their corrosion characteristics. The fuel rod consisted of a 0.020 in. thick core of uranium-aluminum alloy clad with 0.020 in. of 2S aluminum on each side. The assembly was prepared by the regular brazing procedure using 11.5 percent silicon-aluminum brazing alloy and an alcohol slurry of Eutectic 170 flux.

The tests were conducted for 42 days in stagnant demineralized water at 80-85° C. The approximate surface area of each specimen was 620 cm². Data are shown below:

Sample Number	Solution pH	Hydrogen Peroxide	Wt. Gain* mg/dm²	Wt. Loss** mg/dm²	Final Sample Condition
1	*6.1*	*None*	*55*	*35*	*Good condition*
2	*6.1*	*0.0005M*	*50*	*99*	*Pitting on brazed areas*
3	*6.1*	*0.005M*	*17*	*310*	*Pitting on brazed areas*
4	*7.7*	*0.005M*	*278*	*399*	*Severe pitting on brazed areas*

* This weight represents the increase in weight of the specimen as removed from the test solution.
** After defilming.

[redacted]

Figure 31 shows the appearance of the specimens as they were removed from the test solutions. Sample 1 shows the least corrosion product, sample 4 the maximum. These two specimens also showed the minimum and maximum weight losses, respectively, after defilming. The combination of high *p*H (7.5-8.0) and 0.005*M* hydrogen peroxide resulted in the most intensified corrosion attack. This attack in all cases centered primarily on the brazed areas.

Should pinholes or flaws develop at a brazed joint, permitting water to collect into stagnant pools in these areas, the formation of bulky corrosion products would be expected. This is, of course, unlikely under the dynamic conditions of reactor operation. It appears probable that if a surface of uranium-aluminum alloy becomes exposed under operating conditions, the formation of corrosion products at brazed areas will be stimulated, depending upon the condition of the cooling water, and thus afford some galvanic protection to the active material.

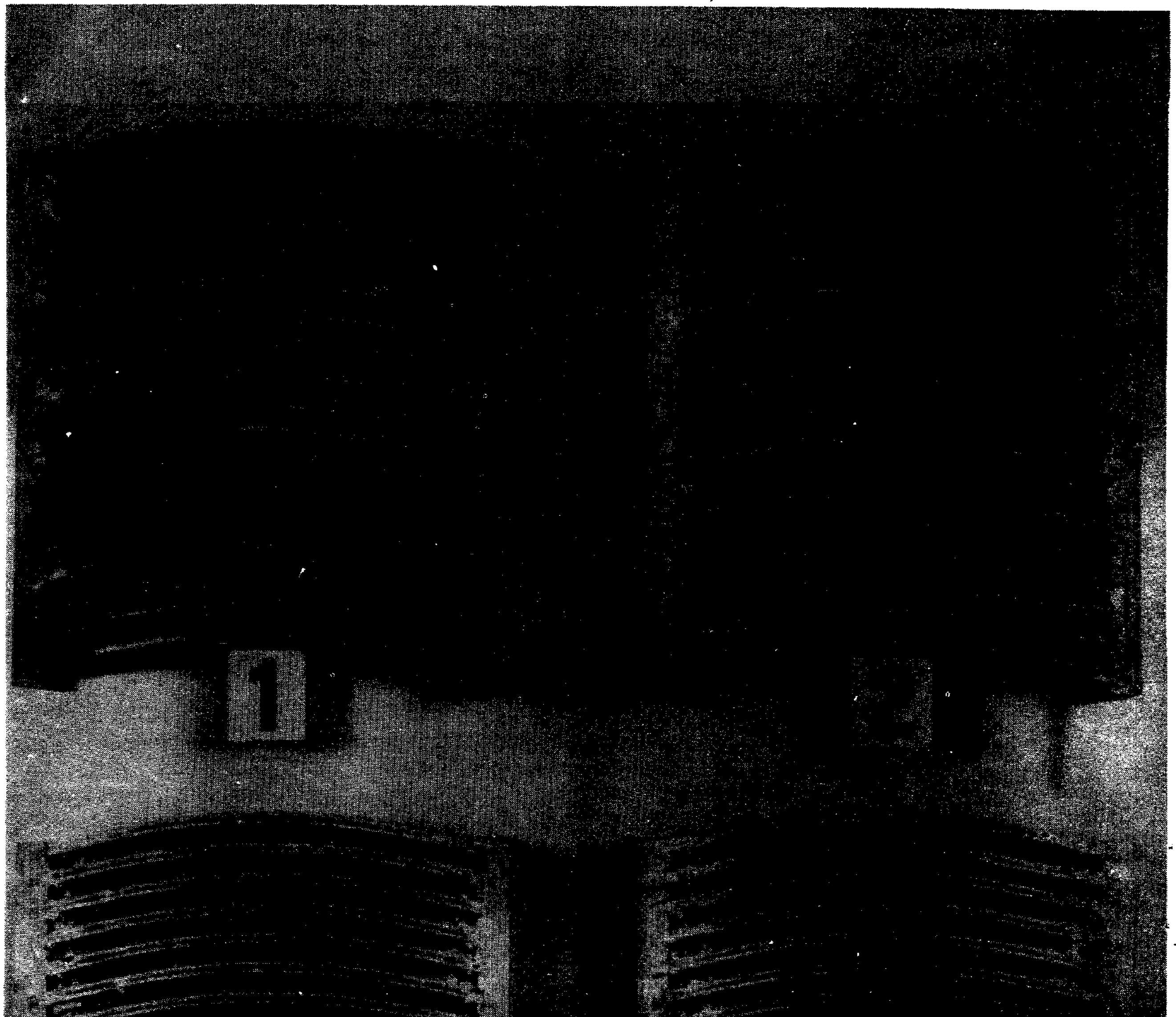
Analyses of the water-soluble uranium in the solutions at the end of the six weeks test showed solution 1 contained 0.2 ppm U; solution 2, 0.5 ppm U; solution 3, 0.5 ppm U; and solution 4, 0.6 ppm U. The figure for solution 4 may be low, since some bulky corrosion products were observed in the bottom of the flask. The uranium concentrations are about those to be expected from direct exposure of the U-Al alloy at these *p*H values.

CORROSION OF TITANIUM AND ZIRCONIUM

Static corrosion tests with titanium and zirconium exposed to simulated reactor water at 85° C were concluded. Mono-metal and galvanic couple tests with 2S aluminum were conducted. The titanium metal was obtained from the Remington Arms Company; the zirconium was produced by the Northwest Electro-Development Laboratory of the Bureau of Mines, Albany, Orgeon, and furnished this Laboratory by the Bureau of Ships.

The purposes of these tests were (1) to study the corrosion behavior of the metals in simulated cooling water for the Materials Testing Reactor, and (2) to compare the galvanic effects of the metals on 2S aluminum, with the galvanic corrosion of the 2S aluminum-347 stainless steel couple. An example of the latter couple in the reactor is the 347 stainless steel spring for positioning the fuel assemblies into the grid.

The test results show that zirconium and titanium have excellent corrosion resistance to the test solutions and that the galvanic corrosion attack on 2S aluminum in contact with either of these metals is decidedly less intense than the attack obtained on 2S aluminum coupled to 347 stainless steel.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3
CORROSION OF ACTI
IN DEMINERAL

Tables 2 and 3 summarize these corrosion data. The tests were of a stagnant type, in demineralized water at 85° C for 60 days. The water *p*H was 5.5-6.5.

CORROSION PRODUCT PICK-UP IN A RECIRCULATING WATER SYSTEM

A recirculating system consisting of a cast iron pump with cast iron impeller, a 2 in. cast iron soil pipe, a square aluminum tank containing 4 beryllium assemblies, an aluminum heat transfer tube, and a stainless steel reservoir and cooler, was operated for 21 days at total recycle, with the water kept at a temperature of 51° C, a *p*H of about 5.5-6.5, and 0.005 *M* hydrogen peroxide. The purpose of this test was to observe the build-up of corrosion products in the recirculating water using no purge water.

The complete results of the test are contained in Table 4 which shows: (1) the steady increase in the concentration of Fe and Al during the first 15 days; (2) the decrease in the water volume caused by various leaks that could not be conveniently stopped; (3) the steady decrease in the specific resistance which was caused by chemical addition to control *p*H and H_2O_2 concentration; and (4) the effect of the sand filter in removing only a small part of the Fe and Al.

This test is not conclusive, for all the cast iron and aluminum surfaces were new and fresh. The fact that during the last five days of the test the Fe and Al concentrations did not increase indicates that the protective corrosion scales may have been nearly completely formed. A new test is being run to determine the need for a preliminary conditioning of new tank and piping surfaces to minimize corrosion and water fouling in recirculating water systems.

This initial test indicated that filtration through sand may not always remove Fe and Al corrosion products as had been indicated by previous tests.

The beryllium assemblies were not visibly affected by this brief exposure to warm water.

HIGH TEMPERATURE CORROSION APPARATUS

In order to obtain corrosion information about metals at elevated temperatures and pressures, a tilting autoclave and necessary auxiliary equipment have been installed for use in tests up to 600° F and 1500 lb/sq in. Large numbers of static tests can be accommodated at one time, and it has been shown by means of a full scale mock-up of a sealed tank arrangement which fits inside the autoclave, that specimens can be tested at flow rates as great as 8 ft/sec. This inner tank arrangement consists of two cylindrical vessels connected by an axial tube in

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TABLE 2

The Corrosion of titanium in water containing hydrogen peroxide at 85°C

MATERIAL	HYDROGEN PEROXIDE	MAX. PIT DEPTH mils	PIT COUNT per cm^2	Wt. GAIN* mg/dm^2	Wt. LOSS** mg/dm^2	PENETRATION RATE mils/month
Ti	0.0005 M	nil	nil	nil	1.1	0.0005
Ti	0.0005 M	nil	nil	nil	0.7	0.0003
Ti	0.005 M	nil	nil	nil	1.1	0.0005
Ti	0.005 M	1.2	0.2	nil	1.1	0.0005
Ti -	0.0005 M	nil	nil	11.2	nil	nil
2S Al	M	7.2	5.4	93.0	33.3	0.024
Ti -	0.0005 M	nil	nil	7.5	1.4	0.0003
2S Al	"	10.0	3.5	80.0	24	0.017
Ti -	0.005 M	nil	nil	9.4	0.7	0.0002
2S Al	"	11.3	4.5	69.0	20.6	0.015
Ti -	0.005 M	nil	nil	4.9	nil	nil
2S Al	"	11.3	8.9	75.0	21.3	0.015

* This weight gain represents the increase in weight of the specimen as removed from the test solution.
** After defilming.

[redacted]

[redacted]

TABLE 3

The Corrosion of zirconium in water containing hydrogen peroxide at 85°

MATERIAL	HYDROGEN PEROXIDE	MAX. PIT DEPTH mils	PIT COUNT per cm^2	Wt. GAIN* mg/dm^2	Wt. LOSS** wt/dm^2	PENETRATION RATE mils/month
Zr	0.0005 M	nil	nil	nil	3.9	0.00012
Zr	0.0005 M	nil	nil	nil	1.9	0.00006
Zr	0.005 M	nil	nil	nil	7.1	0.00022
Zr	0.005 M	nil	nil	nil	4.5	0.00014
Zr -	0.0005 M	nil	nil	neg.	1.9	0.00012
2S Al	0.0005 M	11.6	1.5	51	neg.	wt. increase
Zr -	0.0005 M	nil	nil	neg.	1.3	0.00004
2S Al	0.0005 M	4.0	0.8	48	neg.	wt. increase
Zr -	0.005 M	nil	nil	2.6	neg.	neg.
2S Al	0.005 M	12.0	1.2	38.0	6.5	0.005
Zr -	0.005 M	nil	nil	2.6	neg.	neg.
2 S Al	0.005 M	13.6	2.1	40.5	2.5	0.002

* This weight gain represents the increase in weight of the specimen as removed from the test solution.

** After defilming.

[redacted]

[redacted]

TABLE 4

Corrosion product pick-up in a cast iron and aluminum water recirculating system

HOURS OF OPERATION	GALLONS OF WATER IN SYSTEM (approx.)	Fe, ppm	Al, ppm	SPECIFIC RESISTANCE, ohms	pH
2	440	0.17	0.07	319,000	6.60
26	---	0.50	0.18	149,000	5.76
98	---	0.48	0.21	126,000	5.75
122	---	0.56	0.20	111,000	5.60
146	370	0.56	0.23	102,000	5.60
170	368	0.59	0.27	98,000	5.77
194	362	0.63	0.29	92,300	5.98
218	355	0.61	0.39	73,900	5.43
242	351	0.59	0.30	85,700	5.78
266	345	0.55	0.41	84,000	5.90
290	339	0.62	0.44	82,900	5.94
314	329	0.7	0.3	81,000	6.05
338	321	0.8	0.2	79,000	6.34
*362	314	0.7	0.3	70,000	6.20
Effluent from sand filter		0.11	0.12	89,600	5.13
386	305	0.51	0.34	73,900	6.03
Effluent from sand filter		0.51	0.35	58,200	5.55
410	295	0.53	0.38	68,300	6.15
Effluent from sand filter		0.53	0.38	69,400	6.31
434	287	0.56	0.29	62,200	6.28
Effluent from sand filter		0.55	0.29	67,200	6.32
458	276	0.53	0.29	65,500	6.36
482	265	0.53	0.28	63,300	6.32

* Filter put into operation

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which the test specimens are mounted. Water flows by gravity past the specimens from the upper tank to the lower tank as the autoclave is rotated at regular intervals through 180° C about its center of gravity. Check valves are installed in two other connecting tubes to control the passage of air and steam from one tank to the other. The heating elements and rotating mechanism are controlled automatically by appropriate instrumentation. This equipment was used in the autoclave tests on beryllium.

METALLURGICAL ENGINEERING DEVELOPMENT

The responsibility for continued development of fuel assemblies and beryllium units for the Materials Testing Reactor Project was transferred from the Technical Division to the Metallurgy Division on April 1. Personnel and facilities assigned to this work were transferred at the same time. The following is a summary of the status of the work at the time of the transfer.

FUEL ASSEMBLIES

The mock multi-plate fuel assemblies (using natural rather than enriched uranium in the U-Al cores) required for the reactor mock-up were substantially completed. Subsequent efforts in casting, rolling, and cladding the alloy plates, and fabricating them into assemblies, will be directed toward improving recoveries to minimize recycling, and particularly toward control of blistering of the plates during heating.

Four sets of sample plates of U-Al alloy have been prepared for irradiation at Hanford. In two groups of samples, containing 13% and 30% uranium respectively, the uranium was substantially all U^{238}. A third group contained 30% uranium; isotopic abundance, 38% U^{235}, 62% U^{238}. In the fourth group, the uranium content was 13%; the isotopic abundance, 93.5% U^{235}, 6.5% U^{238}. The processing of these samples before and after irradiation is being done by the ORNL Solid State Group, to determine changes, particularly dimensional, which may be expected to occur in the fuel elements in actual reactor operation.

BERYLLIUM UNITS

Efforts of the group working on beryllium for the Materials Testing Reactor have been directed toward (1) completing the full size beryllium units for the dynamic corrosion testing program, (2) cooperating in the program to establish

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production facilities to be used in forming 3 in. × 3 in. × 40 in. extrusions for the reactor, and (3) developing methods for drilling long (40 in.) water cooling holes in the beryllium pieces. Five full size replaceable beryllium units were completed and are now in dynamic corrosion test.

Arrangements have been made through the New York Operations Office of AEC for the use of a 2750 ton extrusion press at a War Assets Administration-owned plant at Adrian, Michigan. If a short experimental and development program proves successful, the beryllium extrusion requirements for the reactor will be produced by this press.

Experiments indicate that long 3/16 in. diameter holes can be drilled, using *gun barrel* drills with a long narrow channel through the longitudinal axis of the drill shaft. A forced feed coolant system removes chips and heat and permits faster feeding. A feed of one foot per hour is readily attainable.

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CONTROL AND REMOVAL OF AIR-BORNE PARTICLES

A marked increase in air-borne radioactive particle contamination of the Laboratory area was observed in the summer of 1948. Following this, the Technical Division, as a part of a Laboratory-wide program, immediately engaged in the design and installation of remedial measures for certain sources and the evaluation, qualitatively and quantitatively, of all sources. As this latter information was developed, additional design was effected and in some cases carried through to fabrication and installation.

Progress was reported in some detail in ORNL 215 and ORNL 323, Part I. This account will summarize the efforts and conclusions to date.

CLASSIFICATION OF CONTAMINATION SOURCES

Air-borne active particle sources have been separated into five classes:

Class 1--General ventilation air.

Source--Ordinary office and building ventilation.

Activity level--Little probability of particle or gaseous radioactive content.

Method of handling--Normal good heating, ventilating and air-conditioning practice.

Class 2--Laboratory hood exhausts.

Source--*Non-hot* hoods and room ventilation facilities for chemistry and physics laboratories.

Activity level--Low-order contamination well below tolerance, by particle or gaseous radioactive materials.

Method of handling--Normal good ventilating practices to guard against chemical toxicity, with duct work of black iron or mild steel, coated with Amercoat #33 or equivalent.

Class 3--Dry activity-bearing exhaust lines.

Sources--(a) Reactor discharge air, (b) Active metal-working installations, (c) Cell ventilation air--chemical operations, and (d) Counting rooms, hot hoods, etc.

Activity levels--Variable, from source to source and within a single source, from low through moderate to high contamination levels.

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Methods of handling--(a) Designing future hot chemical process equipment to contain activity within the process vessel off-gas lines and reduce the cell air to class 2. (b) Filtration of the air-streams by adequate filters. Certain present installations make use of American Air Filter Company Deep Bed Pocket Filters packed with two layers of FG-50 Fiberglass Filterdown, backed by U.S. Army Chemical Warfare Service Type #6 asbestos base paper filters. Depending on conditions specific to each installation, the filters may be headed by stainless steel electrostatic precipitators. Fans for class 3 service will be located as close as possible to the exhaust stack, so that failure of any part of the system will be safe; i.e., air flow will be *in* and not *out*. Materials of construction for ducts will be the same as for class 2.

Class 4--Wet activity-bearing exhaust lines--low air flow (hot hoods).

Source--Existing laboratory glassware operations (hot) whose off-gas air flow requires pipe only of 1 in. to 2 in. diameter.

Activity levels--Variable; may be high.

Other conditions--May contain water and acid vapors including chlorides.

Methods of handling--(a) Handling facilities for future installations will be included in class 5 installations as far as is possible. (b) Necessary handling of present installations includes the use of type 316 S Cb stainless steel lines (usually concrete or lead-shielded), Type 316 S Cb stainless steel electrostatic precipitators and glass wool (AAF) filters (for precipitator backup) the frames of which should be preferably of stainless steel, or, second choice, of Amercoat #33 painted steel. The applicability of CWS paper filters is uncertain since the paper's resistance to acid vapors has not been extablished.

Class 5--Wet activity-bearing exhaust lines--high air flow.

Source--Process vessel off-gas lines in hot chemical processing areas.

Activity Levels--Variable, may be high.

Other conditions--May contain water and acid vapors, including chlorides.

Methods of handling--Similar to (b) under class 4, methods of handling, except the equipment will be larger in size.

E V A L U A T I O N   O F   C O N T A M I N A T I O N   S O U R C E S

REACTORS

Methods of evaluation--(a) Sampling through K-25 barrier filters, (b) Sampling through CWS paper filters, and (c) Examination and evaluation of filter house dust burden.

Results of evaluations--Based on beta counts taken 24 hours after removal of filters from the reactor filter house, the entire house after a few

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weeks operation reaches an equilibrium activity level of but 500 millicuries of beta emitters. This activity burden is divided equally between the FG-50 first layer and the CWS second layer.

The overall dust burden of the FG-50 layer increases at the rate of about 230 grams a day. A similar value for CWS paper has thus far been impossible to determine, but a top limit of 10 grams per day can be accepted. It is probable that the actual increase in CWS dust burden is about one gram per day. This burden stays on the surface, with little observable penetration.

Under "A" red filtered light, dust penetration entirely through the FG-50 was observed after 44 days; under visual examination, penetration was only 40%. No previous examinations had been made. The great bulk of the dust and the activity retained on the FG-50, was retained in the first one-eighth of the thickness.

In the neighborhood of half of all dust entering the house enters the system with the influent pile cooling air or is generated in the pile itself. The remainder comes largely from the solids contained in the cooling water.

The FG-50 burden contained particles ranging down to sub-micron sizes. The CWS paper burden was observed to contain active particles as large as 120 microns diameter. Radioautographs consistently show large numbers of active particles on the CWS paper specimens.

Over 156 days of operation the uranium content of the filter house increased at a rate of about 0.16 grams per day to reach a value of around 25 grams.

Independent evaluations of similar air-filtering equipment indicate its efficiency is >99.5%. Filter house evaluations have been taken as valid evaluations of the pile activity output, and independent evaluations made by sampling portions of the pile effluent air stream itself support these results.

Treatment of reactor source--The pile effluent air represents a class 3 source of active contaminants. Its handling is, as indicated above, in a filter house between the fans and the reactor, containing FG-50 filter material backed by CWS filter paper. Indications that it is quite effective are supported by Health Physics area surveys.

A direct evaluation across the house itself is in progress.

RaLa OPERATIONS

Methods of evaluation--Sampling through CWS paper filters, headed by an Aerotec cyclone.

Results of evaluation--Two classes of off-gas lines are represented here--class 3 (cell ventilation) and class 5 (dissolver off-gas line and process vessel off-gas line).

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The cell ventilation line has been scanned throughout both 28 and 29 runs. The activity discharged during run 28 was about 15,000 millicuries of gamma active particulate material (reported as one Mev per photon, one photon per disintegration). During run 29 the line output was about 2,600 millicuries. The outputs both were sharply peaked at the sampling period, during which time the product was transferred to the glassware for treatment prior to final evaporation.

The dissolver off-gas line yielded values of 360 and 420 millicuries, respectively.

The vessel off-gas line discharged about 2,200 and 2,500 millicuries, respectively, with the outputs sharply peaked during the sampling period encompassing the transfer to B-6 cell sampling and volume reduction prior to the glass-ware operations.

Treatment of RaLa source--After run 28 a filter building, consisting of two FG-50 layers backed by CWS paper filters, was constructed and installed. During run 29, sampling downstream yielded an output through the filter building of 120 millicuries, indicating an active particle removal efficiency of about 97%. This efficiency represents a minimum, as errors in determinations tend to yield a low value. The confidence belts of the contributing data are sufficiently wide to include an efficiency of > 99.9%.

The present filter building is considered temporary and will give way to a large cell-air decontamination installation, which will service cell air from 706-C also. The proposed installation will be covered in greater detail later.

The two class 5 lines are not yet effectively decontaminated. Prior to run 28 a filter house of FG-50 material was constructed and installed in each line. The line output values reported above were obtained with sampling points beyond the filters. Only rough data are available to estimate the house efficiencies, but a value of 50% is probably optimistic. These class 5 outputs will undergo decontamination in a proposed central installation, designed to handle these lines plus the class 5 lines from 706-C building and from the new 900 area. The installation will be covered in greater detail later.

*706-C **IODINE 131 OPERATIONS***

Method of evaluation--Same as for RaLa operations.

Results of evaluations--Two off-gas lines are represented here--the class 3 cell ventilation line and the class 5 vessel off-gas line. Runs 38 and 39 were completely sampled, yielding value of 14 and 19 for the cell ventilation line, and of 7 and 15 for the vessel line, all values reported as millicuries of gamma active material.

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Treatment of iodine 131 source--The lines are not decontaminated at present. It is planned that in the future both lines will be connected to the proposed decontamination equipment mentioned previously under RaLa operations.

706-C IODINE 135 OPERATIONS

Method of evaluation--Same as for RaLa operations.

Results of evaluations--Again only two off-gas lines service this installation--one class 3 line for cell ventilation, and one class 5 line for all vessel off-gases. Sampling of run 11 yielded discharge values of 170 and 3800 millicuries, cell ventilation and vessel off-gas, respectively, the values again being in terms of gamma active material. The cell ventilation line discharged but 130 millicuries during run 12 despite the fact that twice the material was dissolved. The output was approximately twice, except for one operation usually effected with a number of open pipettings, which were omitted during this run. This result and others, suggest that pipette transfer operations be avoided in future hot process and equipment design.

Treatment of iodine 135 source--Treatment, present and future, is exactly as with iodine 131 operations.

LABORATORIES AND HOODS

Method of evaluation--Sampling through CWS paper filters.

Results of evaluations--Six hoods whose attendent operations were thought most likely to contribute appreciable activity to the hood discharges, have been sampled continuously for over six weeks, crossing several operation cycles. A control sample was taken from the center of the room.

All values obtained to date are extremely low and of similar orders of magnitude. The mean activity levels of the hood discharges and the room air sample are in the neighborhood of 0.001 millicuries of gamma active material discharged per hour. The highest value obtained to date is 0.004 millicuries per hour.

Treatment of laboratory and hood sources--No decontamination is currently applied. In the future, the discharges will be through a system to be described later.

HOT PILOT PLANT--REDOX PROCESS

Method of evaluation--Sampling through CWS paper filters which are headed by Aerotec cyclones.

Results of evaluation--Three off-gas lines service the hot pilot plant -- a class 3 cell ventilation line, a class 5 dissolver off-gas line, and a class 5 vessel off-gas line. Activity discharges from each line have been determined for each of the dissolver charging conditions--no Hanford slugs, 30% Hanford slugs,

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and 100% Hanford slugs. Activity discharges per run on the class 5 lines have been uniformly low (< 10 millicuries of gamma active material) except when improperly aged slugs (less than 4 months) were used. The class B line consistently passed less than 1 millicurie of gamma active material per run. Since future installations, for which the hot pilot plant Redox work was done, call for slugs to be aged > 4 months, such installations should be for low-order sources of contaminating discharges.

Treatment of hot pilot plant Redox operation sources--No decontamination of the off-gas lines is now applied, although all three lines are completely contained within the 205 stack discharge system.

No future decontamination plans are presently contemplated because of the limited extent of the present work.

BUILDING 101 HOODS

No evaluations were made on this installation. The hood discharges represent a class 3 service, since they service the reduction furnaces 1 and 2. A 600 cfm capacity CWS paper filter with suitable duct work, is 30% installed.

PROPOSED INSTALLATIONS FOR AIR HANDLING AND DECONTAMINATION FACILITIES

During this quarter extensive cooperation with the ORNL Engineering and Maintenance Division has resulted in the preparation of plans for installation of new and permanent air handling and decontamination facilities for Buildings 706-C, 706-D and the new 900 Area. Insofar as possible the newly established air handling classification system summarized in the foregoing sections is being used as a standard for this design effort. The recommendations which are being submitted are reviewed in the following sections, and illustrated in Fig. 32.

LABORATORY AND HOOD VENTILATION SYSTEMS--CLASS 2

All duct work for hoods and laboratories in building 706-C and 706-D will be inspected and all leaks repaired prior to setting up a permanent class 2 system to be tied into the new 250 ft. radioisotope stack. All fans in the present system will be removed since new fans are to be installed at the stack. The normal capacity of this system for combined 706-C and D requirements is 45,000 cfm. The new Physics of Solids Building will contribute approximately 10,000 cfm of exhaust air. The hood and laboratory requirements for the new 900 Area are 40,000 cfm. No decontamination facilities will be installed in this system.

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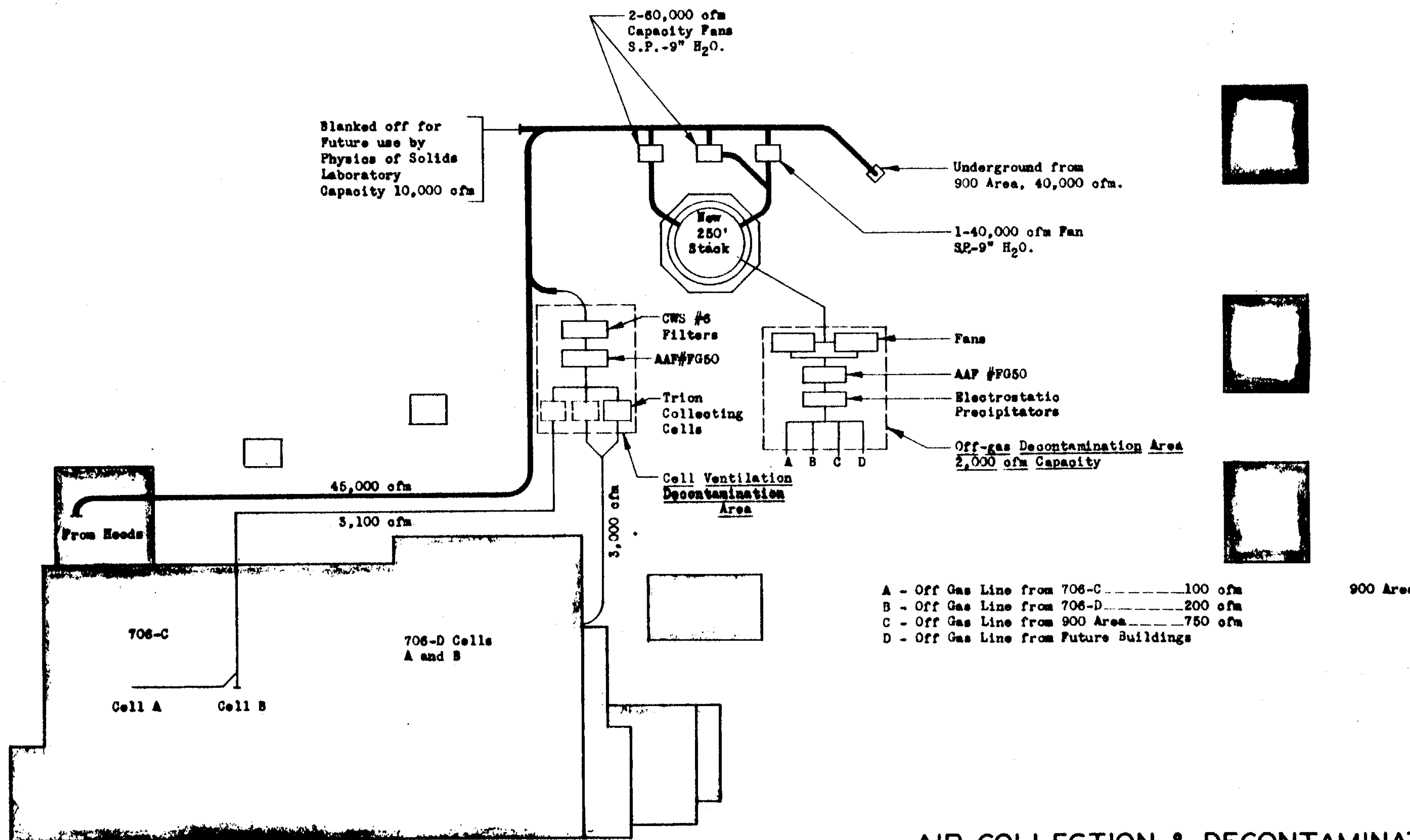

AIR COLLECTION & DECONTAMINATION SYSTEMS-ISOTOPE PRODUCTION AREAS

FIGURE 32

[redacted]

Three fans are contemplated for this class 2 installation. One 60,000 cfm electrically driven fan with a static pressure of 9 in. water gage, will handle the air from hoods and laboratories in building 706-C and D and from the proposed Physics of Solids building. It will also handle the air from the class 3 system for the 706-C and D building cells. One 40,000 cfm electrically driven fan with a 9 in. water gage suction pressure, will handle the air requirements of the new 900 Area buildings. One 60,000 cfm fan with a steam turbine drive will be installed for standby service on any system. This fan will be capable of handling the 900 Area exhaust air or the combined Building 706-C and D hood and cell air requirements. All three fans are to be located just north of the new 250 ft radioisotope stack. No cleaning facility is planned for this air system.

CELL VENTILATION SYSTEM--CLASS 3

The cell off-gas system, a class 3 installation, will have sufficient capacity for handling the combined cell vent air from Buildings 706-C and D. The requirement from these two sources are 3,100 and 3,000 cfm, respectively, a total of 6,100 cfm.

Air decontamination facilities for this system will be required. For the time being the major air cleaning will be done with 7 AAF filters and 11 CWS filters in series. Two 840 cfm Trion electrostatic precipitators will be installed for experimental work in this system. Later a complete bank of precipitators may be installed. The duct work leaving the filters will be connected to the class 2 system before the fans. When completed, this air decontamination system will replace the temporary 706-D cell off-gas filter box at the northeast corner of Building 706-D. Building 706-C cell vent gases will also be cleaned.

HOT VACUUM SYSTEM--CLASS 5

Class 5 air from Buildings 706-C and D and 900 Area equipment will be handled in the Radioisotope Area off-gas decontamination area southeast of the new 250 ft stack. Air ducts to this area will consist of underground stainless steel lines. Four inlet lines to the installation are planned; one from 706-C equipment with 100 cfm capacity, one from the 706-D equipment with 200 cfm capacity, ity, one from the 900 area equipment with 750 cfm capacity, and one spare line for future use.

The air decontamination equipment will include one Cottrell type tube precipitator and 2 AAF filters. The air will be heated slightly just ahead of the

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AAF filters. Two blowers will be installed, one electrically driven for normal use, and one driven by a steam turbine for emergency use. Each blower will have a capacity of 2000 cfm and will have a suction vacuum of 50 in. water gage. The blowers will exhaust directly to the new 250 ft stack.

REACTOR AIR CLEANING

REACTOR AIR INLET ELECTROSTATIC PRECIPITATORS

All design efforts on the electrostatic precipitators for the ORNL reactor inlet air have been curtailed indefinitely. This decision is a result of the good performance of the air filter building. The use of AAF filters for cleaning the reactor inlet air will be continued.

REACTOR AIR FILTER BUILDING FILTER CHANGE

The American Air Filters in cell 4 of the Reactor Air Filter Building were changed on April 25, 1949 with very little difficulty. All work was done from the roof by a small motor crane hoisted to the roof for this purpose. The change, including installation of new filters, was accomplished in eight hours, and it is expected that future changes will take four to six hours. Before the change, an AAF filter had a pressure drop of 4.9 in. of water, as compared with the initial drop of one inch of water.

The filters which were removed were dropped into the filter building canal, and the packing was removed and sealed in boxes, which after draining, were removed to the burial ground. The frames were decontaminated and are now stored in the north end of the canal room. Much of the dust on the filters was washed free in the canal, making visibility in the water so poor, even with lighting, that filter handling operations had to be done blind.

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CHEMICAL ENGINEERING RESEARCH

LIQUID METAL HEAT TRANSFER

The first phase of this problem has been completed and the results are being published in ORNL 361. Four heat exchangers of various lengths and diameters were used, and in all cases the results verified the predictions of Martinelli, Harrison and Menke and those of Lyon, for certain annuli and for tubes, within the limits of experimental error and the knowledge of the physical properties.

Coincident with the move of this group to Y-12, it is planned to rebuild the liquid metal equipment to provide for greater range and flexibility in operation, and to extend the theoretical investigation to annuli with a large ratio of outer diameter to inner diameter.

REACTOR MONITORING

This report will be issued shortly as ORNL-357.

LIQUID METALS HANDBOOK

This proposed handbook is being prepared under the direction of R. N. Lyon as Editor-in-Chief. It will be divided into seven sections or chapters as follows:

Section	*Editor*
Chemical, Physical and Nuclear Properties	R. R. Miller Naval Research Laboratory
Corrosion and Associated Phenomena	L. R. Kelman Argonne National Laboratory
Analytical Methods and Laboratory Technique	C. B. Jackson Mine Safety Appliances Co.
Large Scale Handling	K. D. McMahon General Electric Company
Heat Transfer	R. N. Lyon Oak Ridge National Laboratory

[redacted]

[redacted]

Utilization Not assigned

Availability Not assigned
(Bureau of Mines?)

Development of these chapters is moving on a schedule which should permit publication in October.

COOLANT SURVEY

The results of this survey will soon be issued as ORNL-360. A brief summary of the results is as follows:

1. Consideration should be given to Li^7 as a pile coolant because of markedly superior heat transfer properties of lithium.

2. Time and effort should be expended in developing a process for the separation of Li^7. A reduction in the Li^6 content from 7.5% of natural lithium to 0.1%, would decrease the absorption cross-section for thermal neutrons of the resulting Li^7 and Li^6 mixture to below one barn. This value for the absorption cross-section of Li^7, with 0.1% of Li^6 as an impurity, compares favorably with that of Na or NaK, two liquid metals which are being considered as pile coolants. This corresponds to a removal of about 99% of the Li^6. Complete recovery of Li^7 is not essential however.

3. The heat transfer characteristics and the physical properties of each of the more desirable pile coolants should be ascertained over the complete possible operating range for the coolant.

4. A study should be made to determine the feasibility of producing rare elements such as rubidium and thallium in large quantities at a reasonable cost.

5. The most promising liquid metal coolants, in addition to Li^7, appear to be sodium, sodium-potassium alloy, and lead-bismuth alloy.

6. Little consideration should be given gallium as a heat transfer agent at this time unless the problem of finding a suitable container for gallium is solved.

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RADIATION STABILITY OF MATERIALS

PLASTICS

Several types of plastics have been subjected to different dosages of pile radiation, but the data are still very scattered. Qualitative results of these irradiations are given below. Photographs of most of the materials before and after irradiation are shown in Figs. 33-35. Color transparency photographs are also available for most of these materials. All of the materials were irradiated at a thermal flux ($\sim 0.8 \times 10^{12}$) of about 0.8 of reactor maximum, and at a temperature of about 40° C.

The ultimate aim of the investigation is to establish tolerance dosages of different types of radiation for each material, based on the most important physical properties. From the limited data gathered to date, these materials may be divided roughly into two groups:

1. Those that break down completely within a few days in the reactor--

Methyl methacrylate	Fluorothene
Ethyl cellulose	Teflon
Cellulose acetate	Saran
Cellulose nitrate	Casein

2. Those that will stand up for a week or more in the reactor--

Polyethylene	Phenol formaldehyde
Polystyrene	Nylon

Hardness readings are presented on the basis of the Rockwell R scale for plastics. Impact test results were obtained from standard ASTM test specimens.

Methyl Methacrylate (Fig. 33, Plate I). Twelve hour irradiation produced a very slight yellowing, but no other visual change. Rockwell hardness decreased very slightly and there was a slight decrease in specific gravity. One week of irradiation caused complete breakdown. The specimen swelled and became filled with small holes.

Casein (Fig. 33, Plate II). This material contained a bright red pigment. After twelve hour irradiation a slight fading in color could be detected and the Rockwell hardness decreased very slightly. After one week of irradiation it became a brownish yellow color, was filled with large cracks, and crumbled when handled.

Ethyl cellulose (Fig. 33, Plate III). After twelve hour irradiation, the Rockwell hardness changed from R 113 to R 104; the impact strength changed

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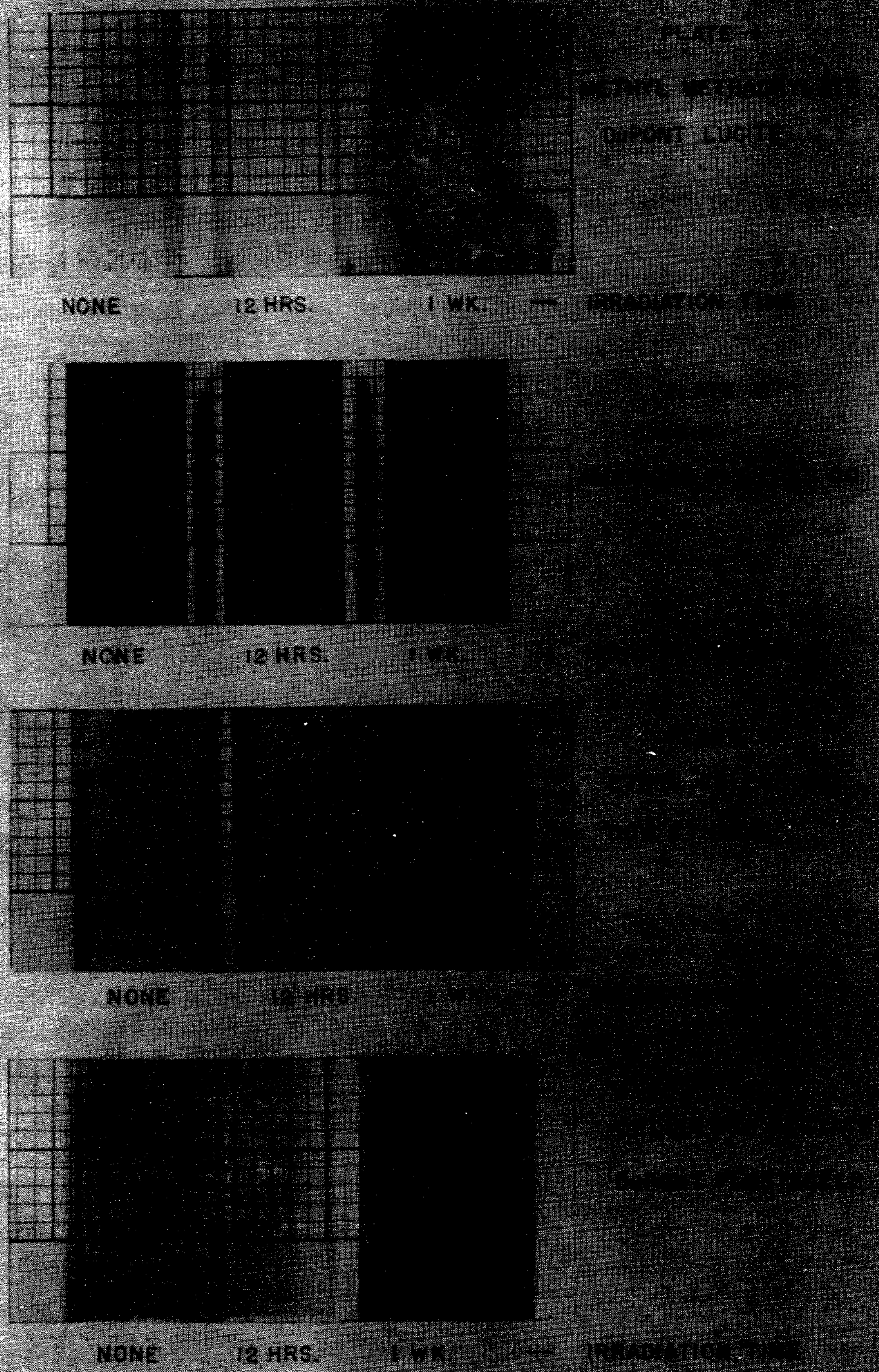

FIG. 33
PLATES I THROUGH IV – IRRADIATION EFFECTS ON PLASTICS

[redacted]

from 2 ft lb/in. to 0.8 ft lb/in. A slight darkening in color was observed. After one week of irradiation the color changed from very bright yellow to dark yellow and the sample crumbled badly.

Cellulose acetate (Fig. 33, Plate IV). Originally transparent with a slight blue tint, after twelve hour exposure it developed a greenish tint and the Rockwell hardness changed from R 103 to R 86. After one week irradiation it became amber colored and crumbled badly. White spots on the 12 hour photograph are hardness test marks. The one week photograph was part of a larger specimen and does not indicate swelling.

Polytetrafluoroethylene (Fig. 34, Plate V). No visible change could be detected after twelve hour irradiation and the only visible change after one week irradiation was that it had lost its glossy sheen. It became so brittle after twelve hours irradiation, however, that it cracked in the hardness machine. After one week irradiation it broke with very slight pressure. The impact strength changed from 3.3 ft lb/in. non-irradiated to 5.7 ft lb/in. after twelve hours irradiation, and to 0.3 ft lb/in. after one week irradiation. All of the deep marks on the photograph are testing and handling marks.

Polyethylene (Fig. 34, Plate VI). Irradiation periods for this sample were one and two weeks. The material changed from a translucent white to a light reddish brown and then to a dark reddish brown. This material showed a marked decrease in ductility and a large increase in hardness. Rockwell hardness went from R -35 to R +32 to R +78. Tensile strength increased slightly for the one week irradiation, but was not measured for the two week irradiation. Impact strength changed from 13.0 ft lb/in. to 3.0 ft lb/in. to 0.8 ft lb/in.

Polyamide (Nylon) (Fig. 34, Plate VII). This material stood up quite well for two weeks exposure. The only large change was in impact strength, which went from 2.8 ft lb/in. to 0.23 ft lb/in. in one week and 0.21 ft lb/in. in two weeks. The hardness changed from Rockwell R 108 to R 114 in two weeks. Tensile strength increased by about 5% after one week irradiation. Percent elongation, which drops very rapidly for most materials, was decreased by a factor of five in one week irradiation. Color, which was originally greenish white, became an increasingly darker reddish brown.

Cellulose nitrate (Fig. 34, Plate VIII). This material is transparent and has a faint bluish tint. After twelve hours irradiation it developed a greenish yellow tint but otherwise showed only slight change in impact strength and hardness. After one week irradiation the specimen became brownish yellow and crumbled to the touch. Round marks on the twelve hour photograph are hardness test marks.

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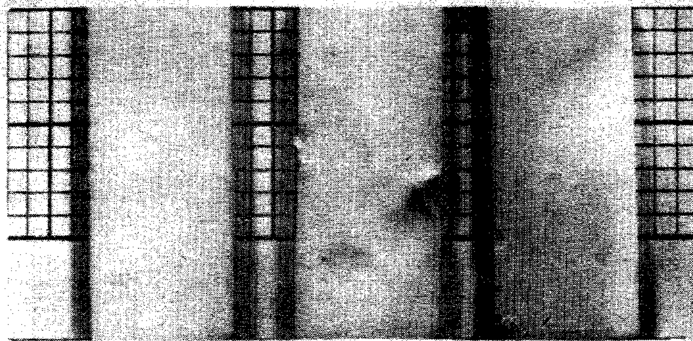

PLATE V

POLYTETRAFLUORO-ETHYLENE

DuPONT TEFLON

NONE 12 HRS. 1 WK. — IRRADIATION TIME

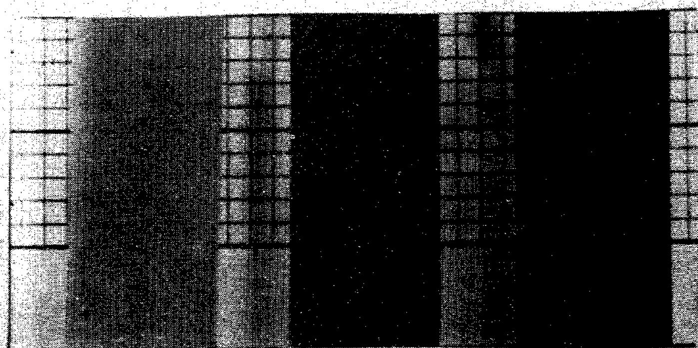

PLATE VI

POLYETHYLENE

DuPONT POLYTHENE

NONE 1 WK. 2 WKS. — IRRADIATION TIME

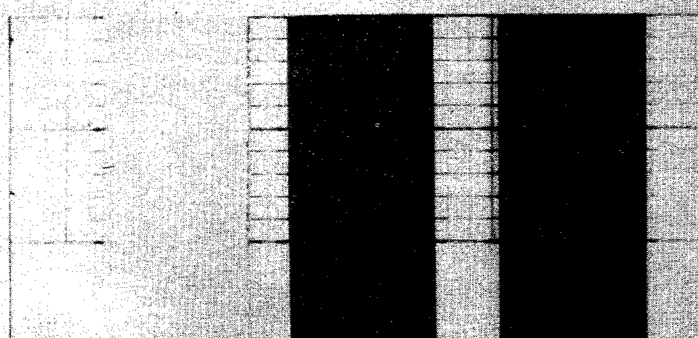

PLATE VII

POLYAMIDE

DuPONT NYLON FM-10001

NONE 1 WK. 2 WKS. — IRRADIATION TIME

PLATE VIII

CELLULOSE NITRATE

DuPONT PYRALIN

NONE 12 HRS. 1 WK. — IRRADIATION TIME

FIG. 34

PLATES V THROUGH VIII - IRRADIATION EFFECTS ON PLASTICS

[redacted]

Polystyrene, clear (Fig. 35, Plate IX). After one week exposure it developed a slight amber tint and a deeper amber color after two weeks irradiation, but remained transparent. To date this appears to be one of the most stable materials tested. Rockwell hardness changed from R 124 to R 121 to R 120. Impact strength changed from 0.22 ft lb/in. to 0.20 ft lb/in. after two weeks irradiation. Tensile strength changed from 4,000 psi to 2,700 psi after one week irradiation. Percent elongation changed from 0.7 to 0.6 after one week irradiation.

Polystyrene, white opaque (Fig. 35, Plate X). This material was originally opaque white. After one week irradiation it darkened to warm gray, and after two weeks irradiation it became a light buff color. Rockwell hardness changed from R 123 to R 121 to R 119. There was about a 10% change in tensile strength and in elongation in one week exposure. No change was observed in impact strength after two weeks exposure.

Polyamide (Nylon) (Fig. 35, Plate XI). This plate shows another type of nylon, similar to Fig. 34, Plate VII. Both types behaved in the same manner. Marks on the photograph are machine marks and not damage.

Linen filled phenol formaldehyde (Fig. 35, Plate XII). This material darkened from light brown to reddish brown in one week exposure, and to dark reddish brown in two weeks exposure. The tensile strength went from 11,000 psi to 4,000 psi in one week exposure, and the percent elongation went from 4 to 0.4 in the same period. Impact strength changed from 2.8 ft lb/in. to 0.25 ft lb/in. in two weeks exposure. Rockwell hardness changed from R 123 to R 120 in two weeks exposure. Percent water absorption changed from 1.6 to 2.6 in two weeks exposure.

Paper filled phenol formaldehyde. This is the normal black Bakelite. A few pin head size blisters and very small cracks were noticed after one and two weeks exposure. Tensile strength changed from 16,600 psi to 3,400 psi in one week irradiation, and percent elongation changed from 2 to 0.2 in the same period. Impact strength changed from 0.6 ft lb/in. to 0.25 ft lb/in. in two weeks irradiation. Rockwell hardness changed from R 123 to R 114 in two weeks exposure. Percent water absorption changed from 1.1 to 7.2 in two weeks irradiation. The irradiated sample, after being subjected to the water absorption test, swelled and cracked badly.

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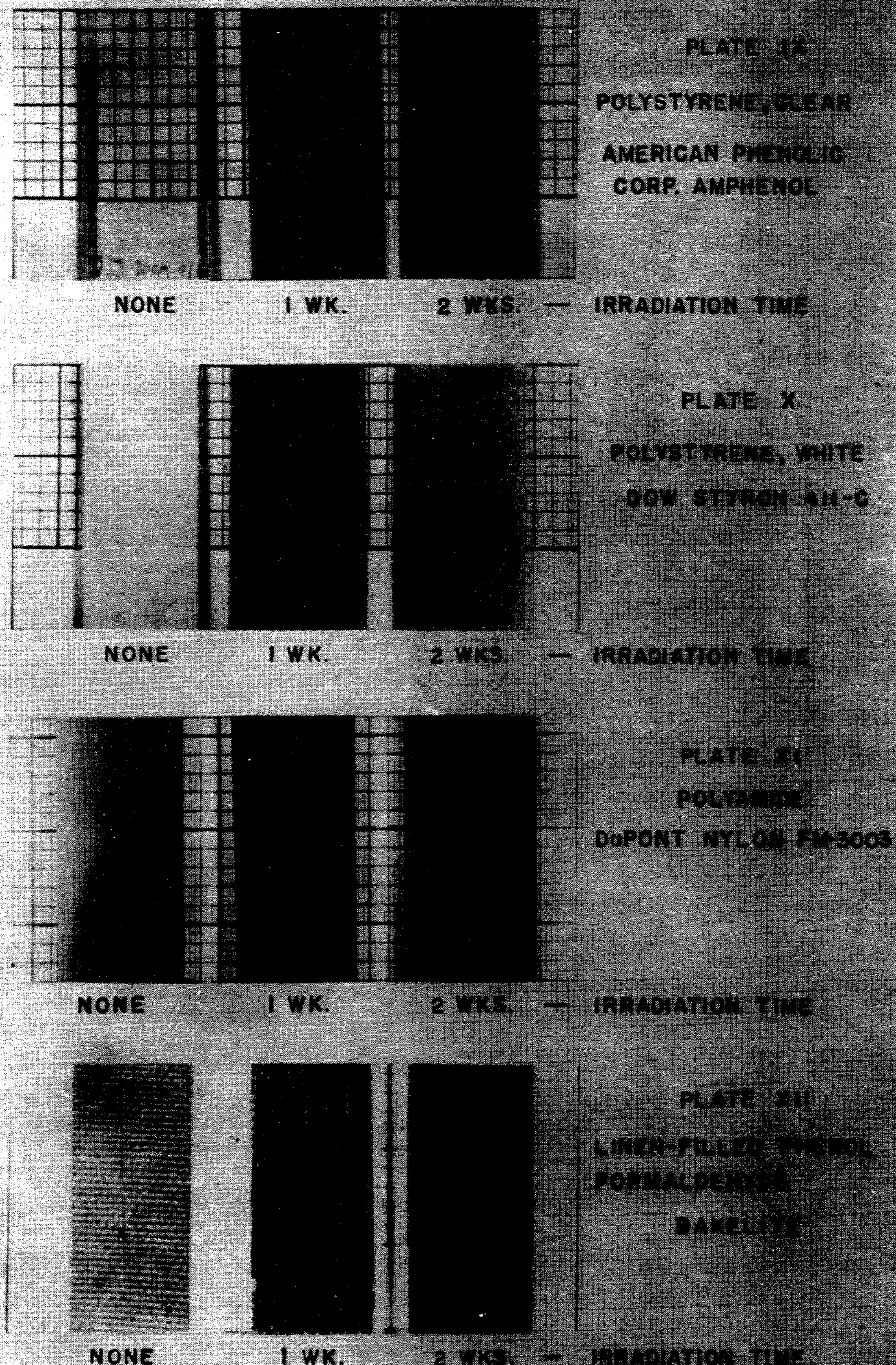

FIG. 35
PLATES IX THROUGH XII - IRRADIATION EFFECTS ON PLASTICS

[redacted]

Asbestos filled phenol formaldehyde. This material is mostly asbestos, with just enough plastic to hold it together. Not too much change was observed in one and two weeks irradiation. Color changed from a light brown to purple, and then to gray-brown. Tensile strength and percent elongation increased slightly in one week irradiation. Impact strength decreased slightly, and Rockwell hardness increased slightly in two weeks irradiation. Water absorption decreased slightly in two weeks irradiation.

Vinylidene chloride--acrylonitrite copolymer (Saran). After one week exposure this material became very soft and flabby and exuded a corrosive liquid. The color changed from translucent brownish-yellow to black. Tensile strength went from 3500 psi to 700 psi. Percent elongation went from 190 to 4. Impact strength went from 1.5 ft lb/in. to 0.2 ft lb/in. Rockwell hardness went from R +72 to R -13.

Polymonochlorotrifluoroethylene (fluorothene). After one week irradiation this material became very brittle and broke up on handling. Color changed from white to light amber tint.

Polystyrene (black). No change in color occurred after one and two weeks exposure. This material contains a black pigment. No change in tensile strength was observed after one week exposure, but the percent elongation changed from 22 to 1. This material will stretch much more than the other polystyrenes tested, probably because it contains more plasticizer than the others. After irradiation the percent elongation was about the same as for the other polystyrenes. Impact strength changed from 0.6 ft lb/in. to 0.2 ft lb/in. in two weeks irradiation. Rockwell hardness changed from R 102 to R 113 in two weeks irradiation.

OILS AND GREASES

Several more points have been obtained to fill in the gassing curves that appeared in ORNL 323, Part I, but no other tests have been started yet on these materials.

GAMMA IRRADIATIONS FOR HANFORD

A request was made by the Hanford Design and Construction Division for tests on the effect of gamma radiation on certain oils and plastics for use in the Redox process. The materials were placed inside a ring of six Hanford slugs. It was hoped that a gamma intensity of 10^5 to 10^6 R/hr would be obtained from this source, but the average intensity over one month irradiation was 1.8×10^4 R/hr. This in-

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tensity is too low to obtain much data in a short period of time. The results of one month irradiation at this flux are given in Table 5. New slugs have been obtained and the radiation intensity has been increased to 1.5 × 10^5 R/hr. The tests will continue for about six months using the stronger source.

REACTOR SHIELD MATERIALS

Metal hydrides are of interest as a reactor shield material. The equilibrium pressure of the more promising of these materials is currently being investigated at NEPA and at their request titanium hydride, zirconium hydride, and tantalum hydride are being tested for gassing under reactor radiation at 40° C. Titanium hydride produced no gas in five weeks in the reactor. Zirconium hydride and tantalum hydride have been subjected to three and one week irradiation periods, respectively, and have produced no gas.

It is now desired that the equilibrium pressure be determined under reactor radiation and temperatures up to 300° C. The experimental equipment for this work is now being planned in cooperation with NEPA. The materials to be tested will probably be the three mentioned and lithium hydride.

IRRADIATION OF EXPLOSIVES

An experimental program is being set up at the request of the Army to determine the effects of gamma radiation on explosives. Work will be done jointly by the Radiation Stability Group of the Technical Division and military personnel familiar with explosive testing methods.

The exposure of explosives to the gamma flux desired (10^6 R/hr) will involve the construction of a totally new apparatus since no appropriate facilities are now available. This apparatus will also serve as a means for exposing plastics and other materials to high intensity beta and gamma radiation. The apparatus will consist of a pneumatic tube arrangement for activating the source, and a shielded setup near the reactor for exposing the samples to the source. It is planned at present that the source will be gold, but other materials may be used if desired.

Design of the apparatus has already begun, and it is estimated that construction and installation will be completed in about six months. Dr. K. S. Warren of Picatinny Arsenal will conduct the tests on explosives.

[redacted]

TABLE 5

Gassing of oils under gamma radiation

SAMPLE	TOTAL RADIATION Roentgen	GAS EVOLVED	REMARKS
DC 550--silicone oil	1.4×10^7	none	no observable change
DC 200--SAB50-silicone oil	1.4×10^7	none	no observable change
DC 200--SAE20-silicone oil	1.4×10^7	none	no observable change
Askarel-Westinghouse 40% trichlorbenzene 60% chlorinated diphenyl	1.4×10^7	none	no observable change
Trichlorbenzene	1.4×10^7	none	no observable change

Results of tests on plastics

MATERIAL	TOTAL RADIATION Roentgen of Gamma	TENSILE STRENGTH psi		HARDNESS, ROCKWELL L SCALE		IMPACT STRENGTH ft lb/in.		WATER ABSORPTION Percent	
		Non-Irrad.	Irrad.	Non-Irrad.	Irrad.	Non-Irrad.	Irrad.	Non-Irrad.	Irrad.
Fluorothene	1.4×10^7	4700	4800	+73	+55	1.94	1.30	.008	.035
Polythene	1.4×10^7	1570	1420	-13	-51	6.79	7.84	.03	.07
Saran	1.4×10^7	--	--	+21	-31	1.68	.94	--	--

MATERIAL	CHANGE IN WEIGHT	REMARKS
Fluorothene	none	slight yellow tinge
Polythene	none	slight yellow tinge
Saran	none	became caramel brown color

[redacted]

[redacted]

SOLVENT EXTRACTION

The laboratory work on the investigation of the relation between the physical properties of solvents and mass transfer in a liquid-liquid dropwise extraction column, has been finished. The correlations are essentially complete and a report covering the problem is in preparation.

Because of the large fraction of extraction occurring during drop formation, it was necessary to separate these end-effects from the overall extraction to analyze the data for extraction during free-fall. This was accomplished by measuring the degree of extraction at various column heights and extrapolation to zero height (Fig. 36). The same diameter was used in a series of experiments employing various solvents in the system solvent-acetic acid-water, with the solvents dispersed. The resulting degree of extraction was then compared with the properties of the solvents employed.

A second series of runs was made to measure the variation of degree of extraction with drop diameter using the solvents and system employed above (Fig. 37). The overall transfer coefficient tends to approach a constant value above a diameter of 0.12-0.14 inches. It is of interest to note that the velocities of the dispersed droplets tend also to approach a constant velocity as one increases the diameter above this range.

To interpret the data covering the variation of the overall transfer coefficient with diameter, a set of runs was made with the system carbon tetrachloride-acetic acid-water, with the carbon tetrachloride as the dispersed phase. The fraction of extraction is plotted as a function of drop diameter in Fig. 38 for a series of runs using a short column and a second series using a long column. The extraction occurring during free-fall can be obtained as the difference. To further clarify the end effects the influence of drop formation time upon percent extraction was determined for this same system (Fig. 39). Drop formation time is not only a measure of the time of contact between the drop and continuous phase while the drop is forming, but is also in part a measure of the degree of agitation within the drop itself while it is being formed. The difference between the series B and series X curves is the percent extraction occurring during free-fall. This difference as expected, is independent of drop formation time.

The mechanism of mass transfer has been correlated on the basis of hydrodynamics and Fick's diffusion law. This has served to fit the experimental re-

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FIG. 36

VARIATION IN LOGARITHMIC MEAN DRIVING FORCE AS A FUNCTION OF HEIGHT

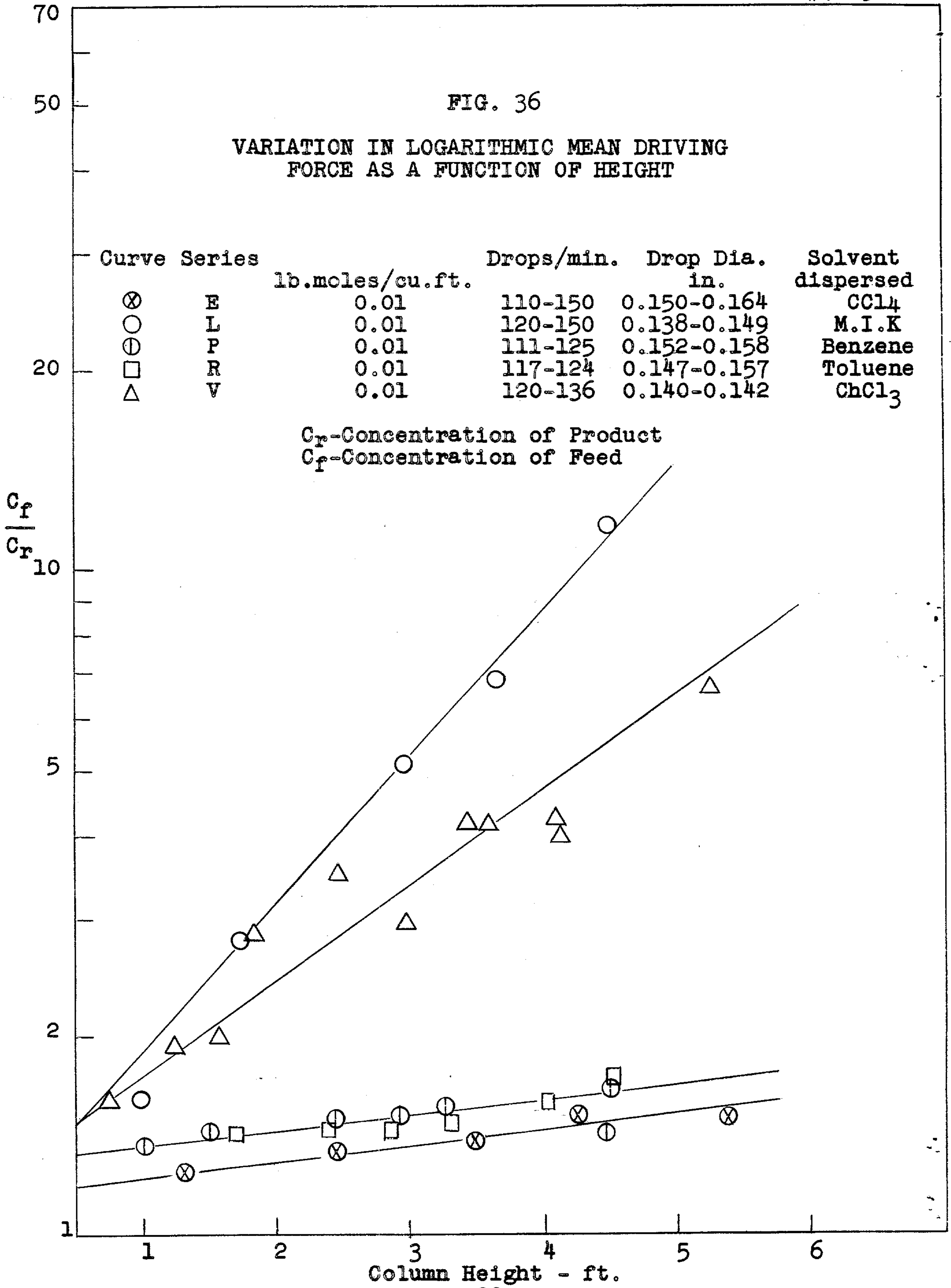
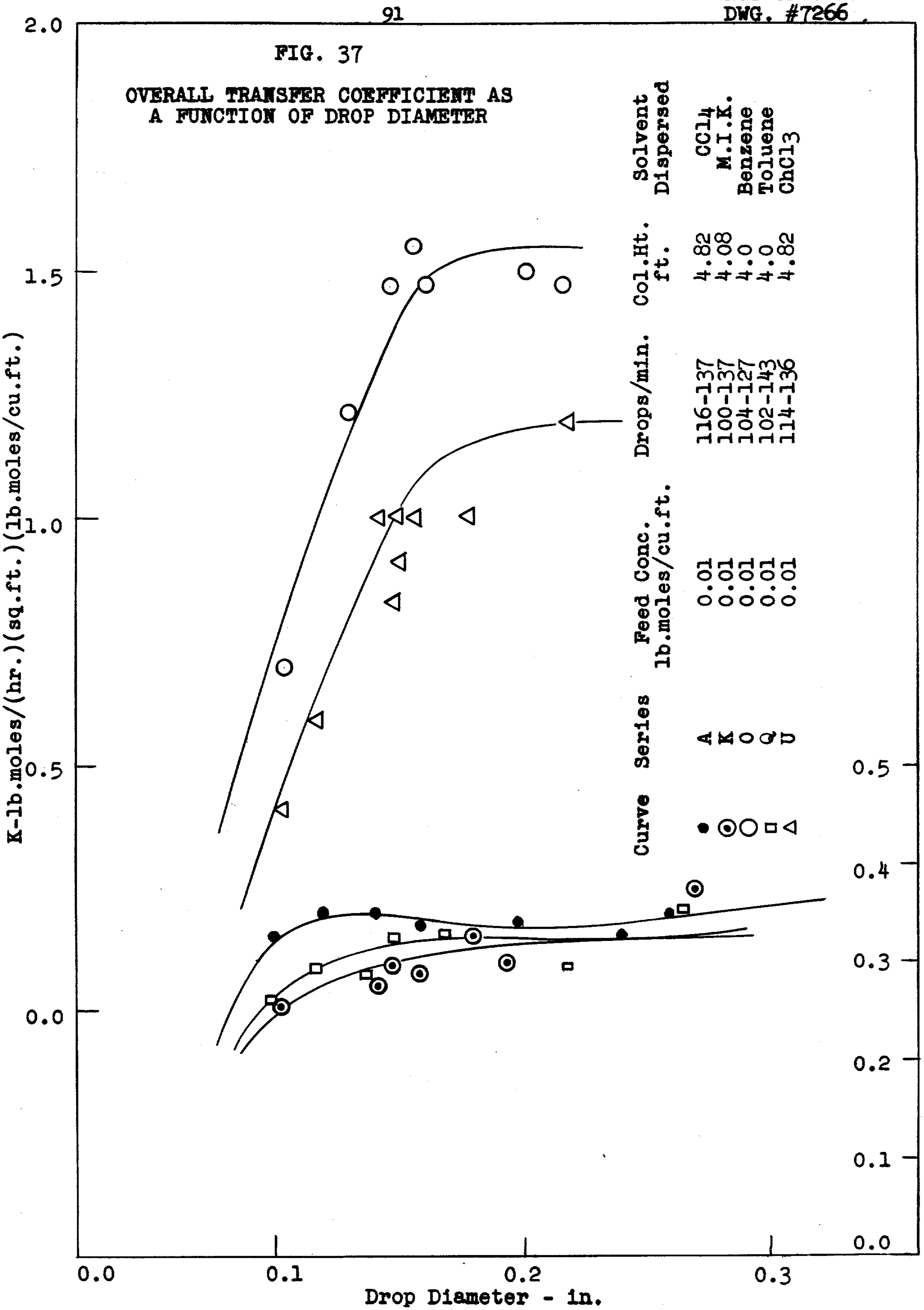


FIG. 37

OVERALL TRANSFER COEFFICIENT AS A FUNCTION OF DROP DIAMETER



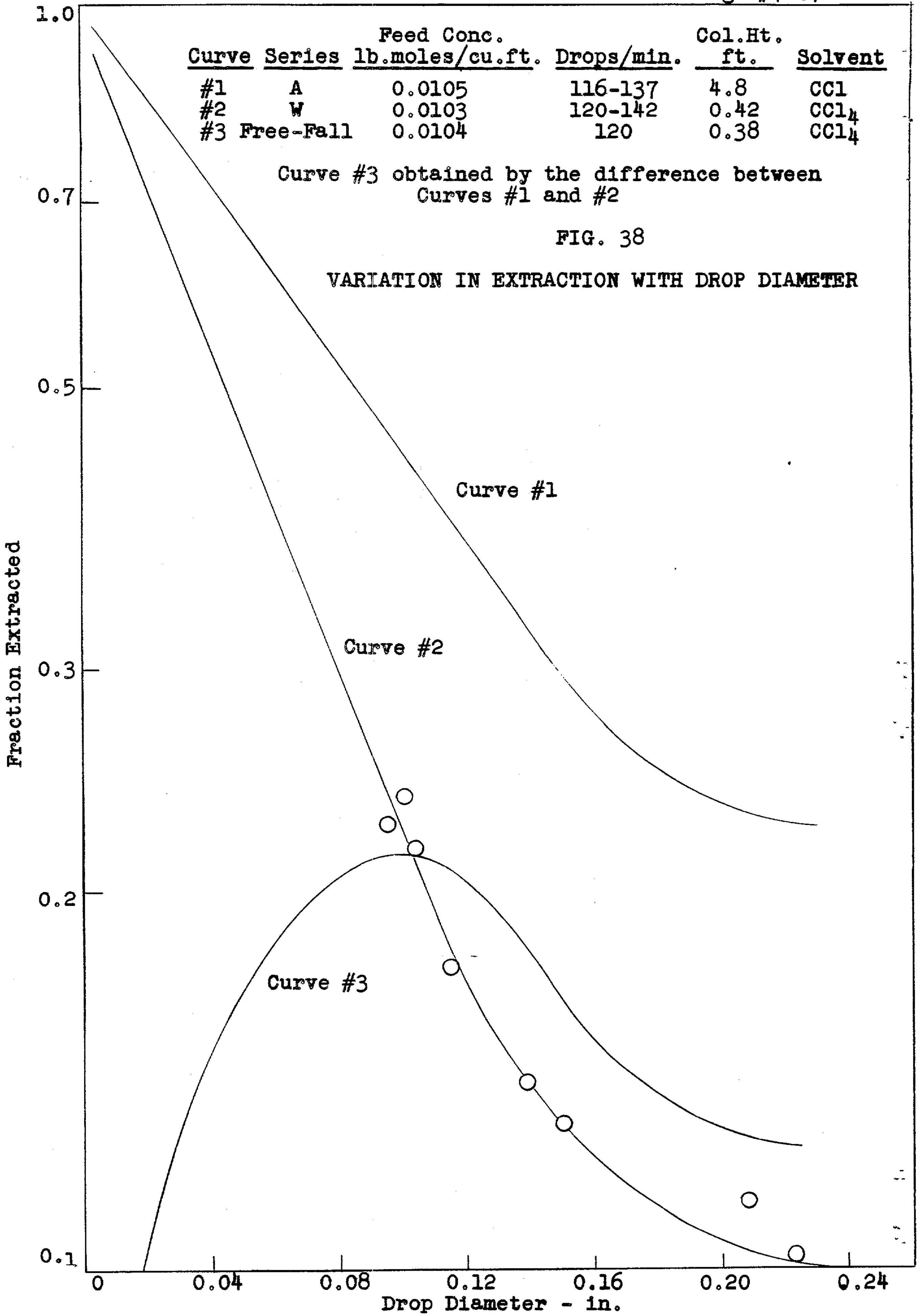
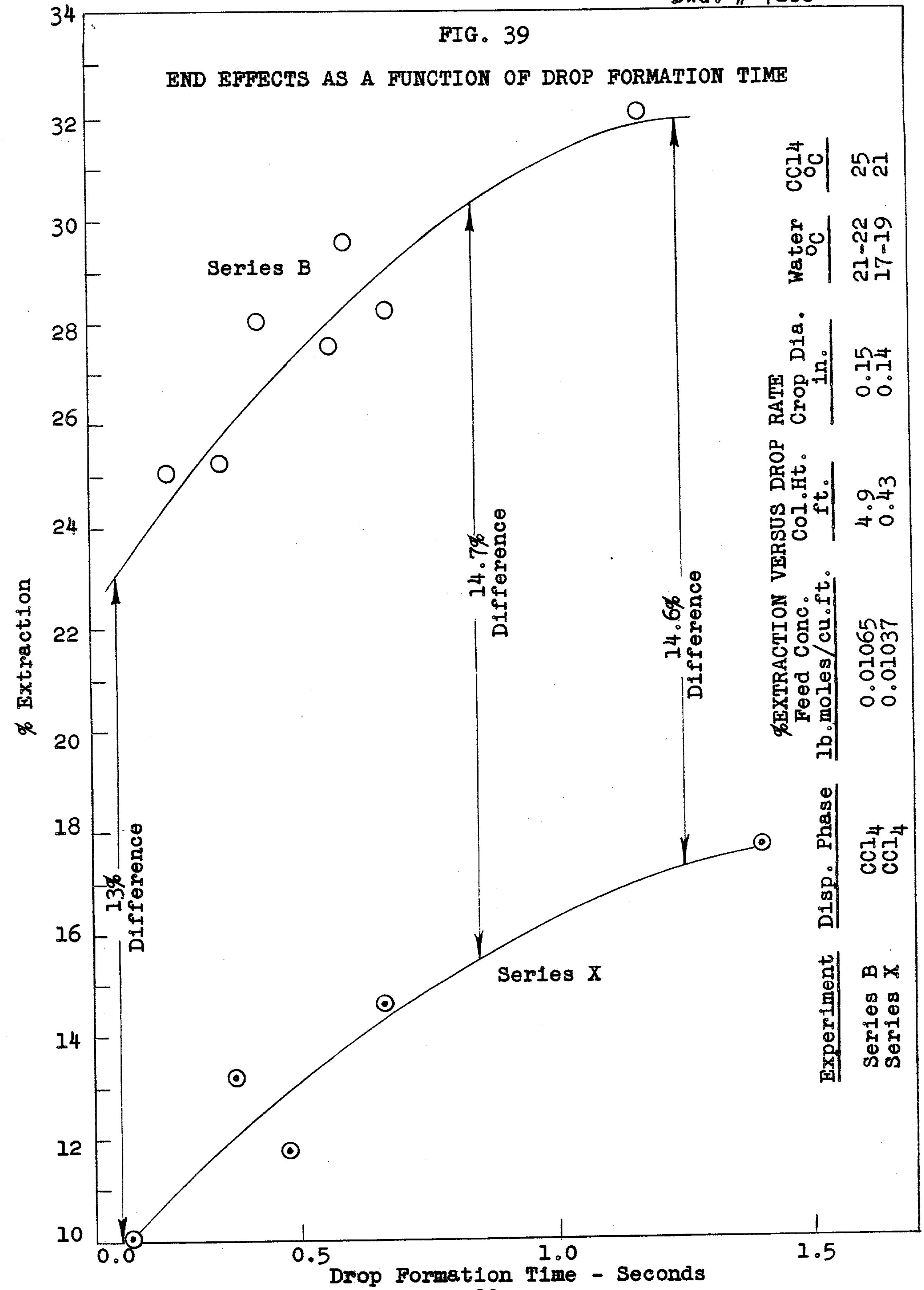


FIG. 38

VARIATION IN EXTRACTION WITH DROP DIAMETER

FIG. 39

END EFFECTS AS A FUNCTION OF DROP FORMATION TIME



% EXTRACTION VERSUS DROP RATE

Experiment	Disp. Phase	Feed Conc. lb.moles/cu.ft.	Col.Ht. ft.	Drop Dia. in.	Water °C	CCl_4 °C
Series B	CCl_4	0.01065	4.9	0.15	21-22	25
Series X	CCl_4	0.01037	0.43	0.14	17-19	21

sults in many instances, but is not totally satisfactory. The results brought out the importance of interfacial tension in extraction, but it could not be determined whether it entered the mechanism through hydrodynamics or surface chemistry. A comparison was made between the results using an overall transfer coefficient based on a fictive film and also using a fictive eddy diffusion coefficient in Fick's law.

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ENGINEERING DESIGN AND DEVELOPMENT

CHEMICAL PUMP FOR RADIOACTIVE SOLUTIONS

The duplex bellows, Flex-O-Pulse timer controlled, solenoid valve type pump, has not been tested due to higher priority programs. As soon as time permits, the pump will be set up and tested in detail.

The pump (Fig. 40) was designed to furnish a means of pumping radioactive solutions at low flow rates, from zero to approximately 30,000 cc/hr, against a head of 50 psi, and with no anticipated leakage from the pump. The valve mechanism is composed of four solenoid valves, two acting in parallel to provide a means of sharp cut off on discharge and suction.

The present pump design incorporates a Bodine drive motor of 50 inch ounces torque at 6 rpm. The actuating drive to the two bellows is composed of a worm gear on the motor shaft which rotates a driven gear that has the shafts coupled to each bellows, threaded through it.

Since during operation one bellows is on suction and the other on the discharge stroke, the motion of the actuating shaft is reversed at suitable intervals to produce the desired flow rate. Reversal is accomplished by means of a Flex-O-Pulse timer which controls the Bodine motor rotation and reversal.

The materials of construction of the pump are stainless steels where contact with the solution is necessary. Two 4¼ in. long, 2 in. O.D. bellows, which permit a ½ in. maximum deflection, are incorporated into the design. Each bellows operates within a stainless steel chamber, with the solution on the outside of the bellows. Bellows were purchased from stock from the Chicago Metal Hose Corporation.

Since previous tests on a similar type bellows have indicated that a bellows operating at half full stroke will have a life expectancy in excess of 15 million flexures, the life of the pump would be based on a similar figure. Nevertheless, since bellows have failed in bellows type valves after only a few hundred flexures, a safety feature to offset this danger has been provided in this pump.

Since a bellow's leak would be toward the inside of the bellows, a shredded Teflon packed gland is provided on the shaft actuating each bellows, in order to contain the leakage.

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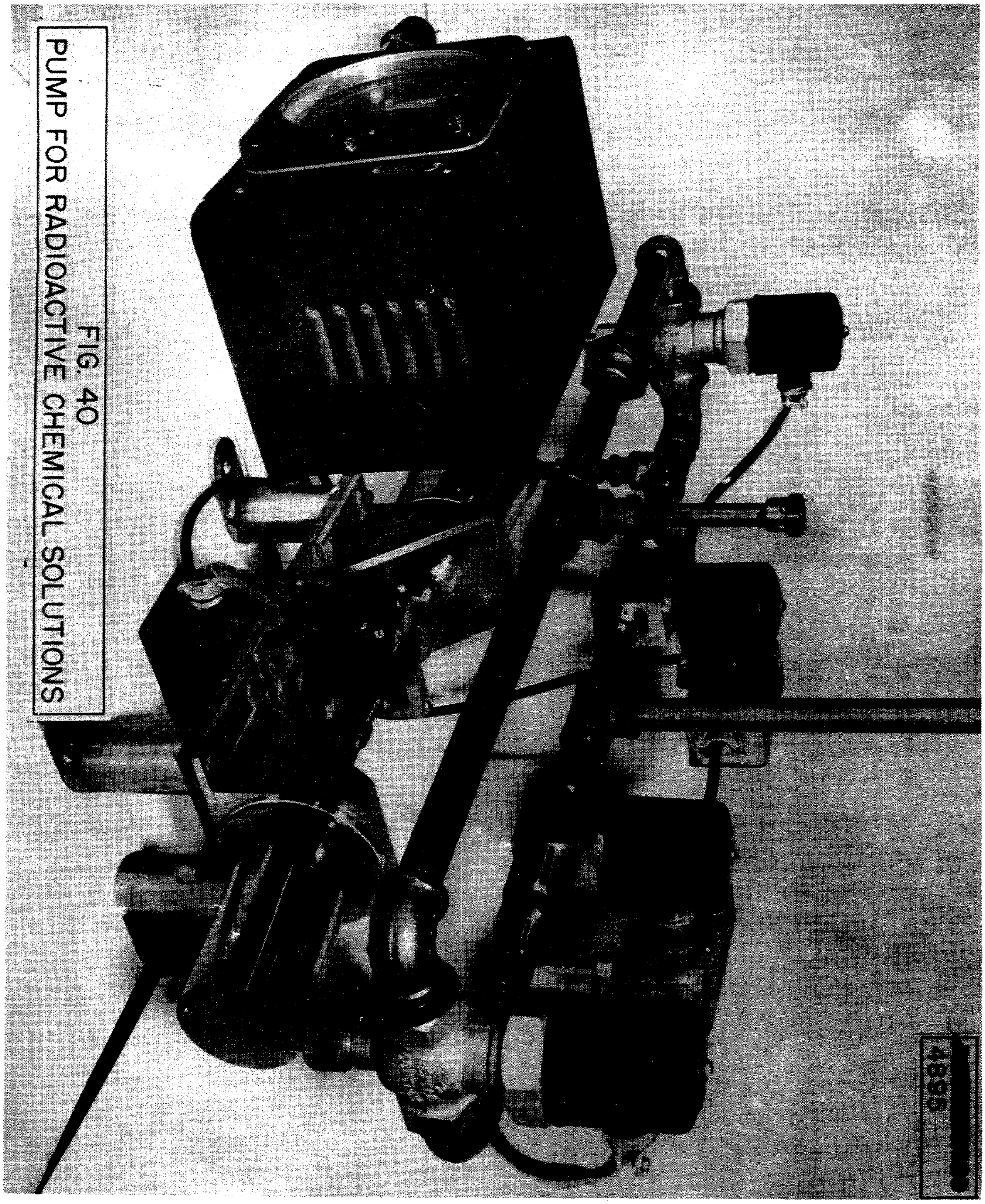

FIG. 40
PUMP FOR RADIOACTIVE CHEMICAL SOLUTIONS

[redacted]

At low flow rates (0-200 cc/hr), the pump strokes may be reduced considerably, thus eliminating in large measure the normal pulsating flow typical of a reciprocating type pump. At higher flow rates the pulsations will be approximately equivalent to the number found in a reciprocating pump.

ISOTOPE LOADER

The chain conveyer type isotope loader for use in ORNL reactor stringer holes is being built and should be completed in four to six weeks. A device for indicating which can is ready for unloading and a shield are being developed and will be ready in this time.

The loader will be supported in an open channel of ¼ in. thick magnesium alloy (97% Mg - 3% Al) plate. The chain rides in an inner channel of 2S aluminum, with the isotope cans carried in rings attached to the chain by riveting them on extended pins. A reversible electric motor with appropriate reduction gears drives the chain at approximately 11 ft per min., or one chain revolution every five minutes.

The present chain will carry 288 standard isotope cans, but there is room in the reactor for a chain that will carry 320 cans. Any future loaders will have this capacity.

Present plans are to unload cans only when the reactor is shut down, but the shield can be modified to permit remote unloading during normal operation.

EXPERIMENTAL SAFETY-SHIM CONTROL ROD FOR THE ORNL GRAPHITE REACTOR

Calculations and design drawings have been completed and a work order has been placed with the shops for the construction of an experimental combined safety-shim control rod for the ORNL reactor to revise the existing control system. This was done at the request of the operating and research divisions of the Laboratory to expand the production and research capacities of the reactor. It is proposed to erect a new balcony adjacent to the north face of the reactor, to facilitate access to certain of the north-south holes. The existing supporting structure for the control rod hydraulic system interferes with this approach. However, by radically altering or even eliminating the mechanism for the shim rod drives in holes 5 and 6, it will be possible to de-

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sign and install the proposed new balcony. By combining the functions of the shim rods with the existing safety rods located in holes 7, 8, 9 and 10 on top of the reactor, this will be possible, and holes 5 and 6 will also be free for other purposes.

The new combined shim-safety rods should be designed for a reactivity equal to the total combined effects of the present shim rods and safety rods to permit elimination of the existing shim rods. Table 6 is presented to show the characteristics of the existing rods and comparison of the proposed new rods with the present rods. The new rods are to have a total combined reactivity of 888 inhours, or 222 inhours each, in order to be equivalent to the existing shim and safety rods.

For isotropically distributed thermal neutrons, the total neutron cross section of a *black* absorber may be obtained from its geometrical dimensions ("Method for Measuring Neutron Absorption Cross Section by the Effect of the Reactivity of a Chain Reacting Pile," H. L. Anderson, et al, *Phys. Rev.*, Vol. 72-No. 1, July, 1947). For a round rod, the equation for this relation is,

$$A = \tfrac{1}{4}\pi D l$$

where,

A = Total cross section, cm^2

D = Diameter of rod, cm

l = Length of rod, cm .

Substituting the dimensions of the existing safety rods in this equation and solving, the total cross section of each rod is found to be 731 cm^2. From the table the reactivity of a safety rod is 105 inhours. Hence, the effect of a safety rod is 6.95 cm^2/Inh. Assuming the new combined shim-safety rods to have the same effect, and knowing the desired reactivity to be 222 Inh, the total cross section of the new rods must be approximately 1530 cm^2 per rod. Therefore, for the same length of rod, the new rod should have an equivalent diameter of 8 cm or 3.14 in. Hence it would appear that a cadmium sheet, 1/16 in. thick and 3 5/8 in. square, would be more than adequate to replace the existing safety and shim rods from a purely absorptional point of view, neglecting the effect on neutron leakage due to the new rod effective diameter.

Based on this reasoning and the limiting dimensional factors imposed by the existing reactor structure, the specifications for the new combined shim-safety rods have been established. The new experimental safety-shim rod is to replace

TABLE 6

Characteristics of control rods of ORNL reactor

	EXISTING				PROPOSED
	REGULATING RODS	SHIM RODS	SAFETY RODS	SAFETY TUBES	EXPERIMENTAL SHIM-SAFETY ROD
NUMBER	2	2	4	1	4
LOCATION	Holes No. 1 and No. 2	Holes No. 5 and No. 6	Holes 7, 8, 9, and 10	Hole 12	Holes 7, 8, 9, and 10
INHOURS (MAX.)	No. 1--185 No. 2--145 330 total	235 each, 470 total	105 each, 420 total		222 each, 888 total
DIRECTION OF TRAVEL	Horizontal	Horizontal	Vertical	Vertical	Vertical
MATERIAL	1.5% Boron Steel	1.5% Boron Steel	1.5% Boron Steel	1.5% Boron Steel	1/16 in. thick Cadmium sheet
SIZE AND SHAPE	1.75 in. sq. bar x 19 ft.--0 in. Lg.	1.75 in. sq. bar x 19 ft.--0 in. Lg.	1.5 in. dia. x 8 ft.--0 in. Lg. rod	5/16 in. dia. pellets	3¾ in. x 3¾ in. sq. rod x 9 ft.--0 in. Lg
METHOD OF OPERATION	Direct electric motor driven	Hydraulic cylinder; normal operation--pump; emergency accumulator	Gravity when latch solenoid de-energized. Out by electric motor	Gravity, when plunger raised from bin outlet by cable manually operated	Normal operation--electric motor. Emergency--gravity when electric clutch de-energized
TRAVEL INTO PILE	Out: To 1 ft.-0 in. outside north face, in shield In: To within 5 ft-0 in. of south face	Out: To 1 ft.-0 in. outside north face, in shield In: To within 5 ft-0 in. of south face	In: 4 ft-0 in. below and 4 ft-0 in. above pile center Out: Just withdrawn from graphite	In: 5 ft.-0 in. below and 12 ft-2 in. above pile center Out: Bin located in top shield	In: 4 ft-6 in. below and 4 ft-6 in. above pile center Out: Dragging top of graphite--12 in.
SPEED OF TRAVEL	No. 1 Max. in 6.9 in./sec. Max. out 6.9 in/sec. No. 2 Max. in 4.76 in./sec. Max. out 4.76 in./sec. Min. in 0.044 in./sec. Min. out 0.043 in./sec.	In: Normal-6 in./sec. Emergency-full travel in 4 sec. Out: 1 in./sec.	About 3 or 4 sec. for full travel	About 19 sec. to fill tube	Normal: 1 in./sec. in or out Emergency: About 1 sec. for full travel
GUIDE TUBE	2¼ in. x 2¼ in. sq. openings cut into graphite blocks	2¼ in. x 2¼ in. sq. openings cut into graphite blocks	2 in. diameter circular openings cut into graphite blocks	1.75 in. O.D. x 0.083 in. wall aluminum tube [illegible]	4 in. x 4 in. sq. opening cut into graphite blocks

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the existing safety rod in hole 7, and is designed for the largest practical dimension which can be inserted in the 4 in. vertical hole in the graphite directly below hole 7. The new rod is to have a maximum overall length of 9 ft, and will have a 3¾ in. square section to provide 1/8 in. clearance from the graphite. The rod will be of sandwich type construction, with the cadmium absorber sheet inserted between an outer sheath of 16-gauge sheet steel and an inner backing of 1/8 in. aluminum plate. This construction was selected to provide adequate support for the cadmium, which by itself would not be self supporting. The rod is designed with the thinnest sheet steel consistent with the necessary rigidity to obtain minimum rod weight. Estimated weight of the rod is calculated to be 95 pounds, as compared with the weight of the existing rod, 48 pounds. A spring shock-absorber has been designed into the rod to protect the graphite in case of over-travel on rapid insertion. It has been necessary to specify a new top plug to accommodate the new rod dimensions and provide adequate radiation shielding. The plug will be fabricated from standard structural steel sections and will be of the same general size and shape as the existing plug. Radiation shielding is obtained by pouring concrete, of the same mix as the concrete shielding presently surrounding the reactor, into the annulus between the outer sheath and the inner 4 in. square entrance tube.

Before proceeding with the replacement of safety rods in holes 8, 9, and 10 and the elimination of the existing shim rods, the new experimental rod is to be calibrated against the existing regulating rods, using data and methods presented in BNL-22, "Calibration of Control Rod System of the X-10 Reactor," by J. S. Levin and J. W. Weil.

HOMOGENEOUS REACTOR CORROSION PROGRAM

A corrosion program for the proposed homogeneous reactor is in progress. The immediate objective is to select, for more thorough study, materials which show promise of good corrosion resistance in aqueous uranyl sulfate solutions containing 3.0-3.5 gm of uranium per liter at 750-1500 psi pressure and 150-350° C. Preliminary tests are being conducted at 100° C and atmospheric pressure. Tests will be conducted also in sealed silica glass capsules at 200-250° C. These exploratory tests are planned to aid in the selection of suitable materials for construction of an autoclave or autoclave liner, in which tests may be conducted under proposed operating conditions of the homogeneous reactor.

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High purity uranyl sulfate $UO_2SO_4 \cdot 3H_2O$, containing 57.5 percent uranium and free from nitrates, is used. The following spectrographic analysis was obtained from Y-12:

Ag	1 ppm	Fe	60 ppm	Sb	2 ppm
Al	80 ppm	Mg	40 ppm	Si	1000 ppm
Cd	2 ppm	Mn	1 ppm	Zr	150 ppm
Cu	4 ppm	Pb	450 ppm		

The *p*H, 3.35, of a 0.0103 *M* solution of this uranyl sulfate compared favorably with the *p*H, 3.56, of a known high purity uranyl sulfate solution of similar concentration as reported by Helmholz and Friedlander (LAMS-30), indicating the absence of appreciable amounts of oxides and basic salts.

It was planned to add hydrogen peroxide to the preliminary test solutions to more closely simulate the effects of irradiation on the uranyl sulfate solutions. However, hydrogen peroxide caused the immediate formation of insoluble uranyl peroxide. The presence of this precipitate was undesirable for test purposes, so the peroxide addition was eliminated. It has been previously reported (M-CC-1323) from another installation, that small additions of sulfuric acid will greatly increase the amount of hydrogen peroxide necessary to cause precipitation of uranyl peroxide from uranyl sulfate solutions.

Tests were made to determine the concentration of sulfuric acid required to prevent the formation of UO_4 by hydrogen peroxide in solutions having the uranium concentration tentatively specified for the homogeneous reactor. Various quantities of one molar sulfuric acid were added to 50 ml samples of 0.0147 *M* uranyl sulfate. These samples were then titrated with 1.44 *M* hydrogen peroxide to the point where precipitation was first noticed on standing. The results are shown in Table 7.

Seventeen different metals and alloys are currently under exposure to uranyl sulfate solution containing 3.5 gm of uranium per liter. The tests are operated under stagnant conditions at 100° C.

The metals and alloys in test include: palladium, rhodium, gold, silver, platinum, titanium, nickel, Ticonium III, Illium, Carpenter 20, tantalum, Ni-Resist, zirconium, Hastelloy A, Hastelloy B, Hastelloy C, and Hastelloy D.

Nickel, Ni-Resist, Hastelloy Alloys A, B, and D, all show signs of severe corrosion attack. The other materials show no visible evidence of attack. The tests will be continued for 1000 hours and the results will be included in a subsequent report.

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TABLE 7

Precipitation of UO_4 from uranyl sulfate solutions with hydrogen peroxide

Ml OF URANYL	MOLARITY OF URANYL SULFATE	MOLARITY OF H_2SO_4	*pH* OF SOLUTION	MOLARITY OF H_2O_2 REQUIRED FOR UO_4 PRECIPITATION
50	0.0147	0.00	3.20	0.0014
50	0.0147	0.02	1.64	0.014
50	0.0147	0.04	1.44	0.38
50	0.0147	0.06	1.36	0.98
50	0.0147	0.08	1.28	1.15
50	0.0147	0.10	1.23	1.35
50	0.0147	0.12	1.16	1.64
50	0.0147	0.14	1.10	3.46
50	0.0147	0.16	1.10	3.5 (no ppt.)
50	0.0147	0.18	1.08	3.5 (no ppt.)

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MODIFICATION OF BUILDING 9204-1 AT Y-12

The modification of Building 9204-1 has proceeded satisfactorily during this period, and should be completed on the scheduled date of July 1. The corrosion group of Section IV will begin setting up its operating equipment shortly after June 1, since that phase of the modification was planned for completion first. The completion of facilities to be used for material testing by Section V, including the radiation laboratory and counting room, follows the completion of the facilities for the corrosion group. The Engineering Development Group, Section II, will follow.

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SHIELDING

LID TANK ATTENUATION TEST FACILITY

The ORNL lid tank facility is used to measure the attenuation, through shielding samples, of radiation typical to a chain reacting pile, in an effort to determine the most efficient type of shield with respect to materials cost, weight, and thickness, for any particular installation. The facility consists of a large water tank adjacent to the shield of the ORNL reactor. A rectangular hole, 28 in. × 32 in., extends through the west, or discharge side of the reactor shield, the axis of the hole lying on the horizontal center line of the pile. Radiations emerging from the hole consist of a thermal neutron flux of approximately 10^7, and a gamma beam of approximately 10^3 r per hour. It is hoped that, because of the low background in the tank, neutron and gamma attenuations of 10^9 and 10^6, respectively, can be measured. Since the fast neutron flux is two orders of magnitude below that of the thermal flux, a uranium plate is placed between the tank and the shield, covering the hole. This source plate converts approximately 90 percent of the thermal neutrons that strike it into fissions, which in turn produce the fast neutron flux of known spectrum or energy distribution, required for typical shielding measurements. The samples to be tested are placed in the tank and have approximately twice the area of the source, to afford an interpretable geometry. The samples are built up of thin slabs of material which are added successively to the tank while the radiation transmitted by different thicknesses is measured in the surrounding volume of water.

A tank containing water is used because the water is a good neutron shield and reduces radiation hazards to personnel. At the same time it allows convenient access for insertion, motion, and removal of detection instruments and sample slabs, without requiring a lengthy reactor shutdown. Since the water is a relatively poor gamma shield, the gamma flux can be measured without the interference of neutron-induced activities in the ion chambers, etc., by maintaining a foot or more of water between the samples and the gamma detecting instruments to filter out the neutrons. The tank is several feet larger than the samples and reduces the neutron background well below that of the general level in the operating area, this background being due to neutrons emerging from the reactor shield itself.

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CONSTRUCTION FEATURES

The tank is constructed of ¼ in. mild steel plate. The internal dimensions are 7 ft × 7 ft × 11 ft, the longest dimension being horizontal and parallel to the shield face. The tank rests on two 14 in. I-beams connected by three 6 in. beams which support the samples. These were necessary to avoid overloading the floor, since the samples are very dense. The tank is filled with water to within 6 in. of the top and holds approximately 30,000 pounds. A 4 in. water line has been provided to fill the tank rapidly with filtered water, and a continuous flow of water through the tank is maintained by means of an overflow just below the top. The tank is surrounded on three sides by two feet of concrete in the form of small blocks to provide additional gamma shielding and to reduce the gamma background for low level measurements in the tank.

The hole which extends through the shield is 28 in. × 32 in. at the outside, and contains three steps which reduce the opening facing the pile to 23½ in. × 27½ in. An aluminum tank completely fills the hole and can be rapidly filled with water or drained; this acts as a radiation valve, permitting the source to be turned on or off at will. A source container is placed in a 3 in. recess in the face of the tank adjacent to the pile shield. The source consists of uranium slugs of the same type as used in the Clinton pile. These slugs can be removed individually from the source container, thus reducing radiation hazards attendant upon removal of the source from the tank. Three boron shutters are provided to control further the radiation emitted by the source.

The shutters were constructed by mixing nine parts of boron carbide with one part of shellac and three parts of thinner, and spraying this mixture as a paint on two sheets of 16 gage aluminum to a thickness of approximately 3/16 in. The aluminum sheets were then fastened to a steel frame so that the boron carbide would be contained and contamination problems reduced. The first shutter is a 32 in. × 36 in. rectangle and can be interposed between the source and the thermal neutron flux emerging from the hole. This will reduce the fast neutron flux from the source by a factor of 100 and hence allow background measurements, the background consisting of radiations from the pile rather than from the source. The second shutter is also interposed between the thermal flux from the pile and the source, and contains circular holes of various diameters to permit different geometries. The third shutter is placed between the external tank and the source to prevent variations of the source strength due to thermal neutrons which have been scattered back into the source from the tank itself.

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SAMPLE HANDLING

Since the samples each weigh in the neighborhood of a ton, it is necessary to supply hoists to transport them into the operating area and place them in the tank. The samples themselves are 56 in. × 66 in. × 1-7/8 in. maximum, the thicknes being limited by the one ton capacity of the overhead hoist. Samples such as iron, lead, and tungsten, which will not absorb water, are positioned in the tank by means of steel racks with slotted sides. Five such racks, each handling a one foot thickness of sample, are necessary to keep the volume behind thin shielding samples free from obstructions which may impede movement of the detecting instruments. Samples which would have their shielding characteristics changed by the absorption of water will be placed in water-tight tanks. These are to be built with a minimum amount of metal in the faces perpendicular to the flux, and will be approximately 1 ft × 6 ft × 7 ft. A plastic bag which can be filled with water will be placed inside the tank. This is necessary to fill up any space in the tank not filled by samples, since discontinuities in the medium surrounding the samples will cause difficulties in the interpretation of the measurements.

INSTRUMENTATION

Initial measurements have indicated that, to obtain all the data required for gamma transmission calculations, a method must be found to increase the speed of such measurements. This is due to the fact that a complete integration of the gamma flux passing over one Z-plane (perpendicular to direction of attenuation) requires something like 400 individual readings with an ion chamber. This chamber, being very sensitive, has a time constant requiring five minutes for each reading or four 8 hour shifts for the measurement of one Z-plane. Since attenuation measurement for each slab in a shield requires two or more integrations, a time saving in this operation will speed up the entire program.

A method has been developed to reduce this time by changing from a hand operation to a motor driven servo-operation.

The detecting instruments are mounted on a carriage which provides a three dimensional movement within the tank. The horizontal movement (X-coordinate) is driven by a servo type motor, actuated by a standard Brown amplifier with a high impedance input. The vertical movement (Y-coordinate) is driven by a reversible variable speed motor. The movement (Z-coordinate) perpendicular to the source still is hand operated. Gamma measurements are now taken by placing a chamber at a fixed Z plane and setting a voltage proportional to the desired intensity into the electrometer. If the signal from the chamber does not balance

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the set voltage, the servo is actuated, to drive the chamber horizontally toward the center plane of the tank if the signal is too low, and in the reverse direction if it is too high. The chamber moves until a balance point is reached, then oscillates a few centimeters about the null point. The vertical drive motor is then actuated, causing the chamber to move in a region of constant flux through the tank. The X and Y coordinates of the chamber are continuously plotted by means of a two dimensional Brown recorder. By setting some ten to twelve different intensities into the electrometer a complete flux contour map of any given Z plane results. A complete integration of the flux through such a Z plane can be obtained by measuring with a planimeter the area which each isogram encloses, and plotting this against the intensity for each isogram. The area under this curve is then the required integration. The time required for mapping a Z plane is of the order of three hours, whereas to obtain the same information by hand operation would require several eight hour shifts.

ENGINEERING TESTS ON MO

The Knolls Atomic Power Laboratory has tentatively selected MO (a magnesium oxychloride cement with iron aggregate) for parts of the shield in the intermediate reactor, particularly the thermal shield and the top section enclosing the control rods. Meetings in Schenectady with the design, construction, and theoretical groups cleared up most of the remaining questions and a detailed report is forthcoming. The work this period has consisted mainly of tests on thermal conductivity and stability, strength of structural members, techniques of large-scale pours, and expansion problems.

The Materials Testing Reactor group is cooperating in tests on reinforced beams, MO-Portland cement concrete interface characteristics, and differential expansion tests with Portland cement concrete and MO. Results to date appear very favorable.

An arrangement has been developed to bake out water from MO test slabs equivalent to that driven out after several weeks at 300° F. This water re-enters very slowly, only a few percent per day, and it is felt that this technique will be satisfactory for studying radiation attenuation through MO at simulated elevated temperatures. Tests on water diffusion through standard coatings have been disappointing.

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INEXPENSIVE HIGH DENSITY AGGREGATES

Cost of materials for MO, using steel punchings and chilled iron shot as presently specified, runs approximately $360 per cubic yard, of which well over $300 is for the 4½ tons of aggregate. It is hoped that a suitable aggregate may be found which will reduce this cost (at not too great a density decrease) to something nearer $100 per cubic yard, making it economically competitive with Portland cement concrete. Several promising materials are being studied, including a didymium carbonate byproduct suggested by Hanford.

BORAL

Techniques are being developed to cast, roll, and clad quarter-inch sheets of Boral, a 50-50 (mol per cent) mixture of granular B_4C in a metallic aluminum matrix. Sheets up to 20 in. square can be made with available equipment, and negotiations are in progress to use equipment which will produce sheets up to six feet square. Thermal conductivity of boral appears to be as good as steel, and the tensile strength is adequate for proposed shielding applications. Exact figures will be included in a forthcoming report. Boral has been successfully sawed, sheared, punched, drilled, welded, ground, clad with aluminum, and hot rolled. Punch shear strength tests will be made. The efficiency of boral as a thermal neutron shutter is being studied.

MISCELLANEOUS SHIELDING WORK

Members of the shielding group are participating in the project-wide preparation and editing of a group of papers covering the present status of the field of shielding. It is expected that these papers will be declassified and published in the open literature.

T. Rockwell has represented the Division in a meeting of the AEC Shielding Group at Brookhaven on March 14-15. This group is concerned with the coordination of shielding programs at different installations, and with making recommendations concerning declassification standards for the shielding field.

Within the Laboratory the group is furnishing assistance in the shielding problems of the new hot laboratory facilities and of the isotope group.

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PERSONNEL

The chart, Table 8, lists the persons whose work is represented by the material in this report.

TABLE 8

Technical division personnel

M. D. PETERSON - Director — *Louise Bond*, Secretary*

- **W. M. HAWKINS, Assistant Director - Administration** — *Jeanne Doran, Secretary*
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      - Arlene Kibbey
      - C. V. Ellison - Metal Recovery
      - D. E. Ferguson - Metal Recovery
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